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REPORT NO. IDO-28506

AEC RESEARCH AND DEVELOPMENT REPORT

C-83 "REACTORS-SPECIAL FEATURES OF MILITARY PACKAGE POWER REACTORS"

(M-3679, 22ND ED.)

CONTRACT AT (10-1)-880

MASTER

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

GCRE-I HAZARD SUMMARY REPORT

Classification cancelled (or changed to) UNCLASSIFIED
by authority of Naval Sr. Adv. T. Murphy, IDO dated 7-8-59
by J. C. Ridenour TISOR, date 12-18-59

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DECEMBER 1958

PREPARED BY
MILITARY PROJECTS DIVISION

306 001

Aerojet-General Nucleonics

SAN RAMON, CALIFORNIA

A SUBSIDIARY OF THE AEROJET-GENERAL CORPORATION

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Report No. IDO-28506

GAS-COOLED REACTOR EXPERIMENT-I

HAZARD SUMMARY REPORT*

ABSTRACT

The Gas-Cooled Reactor Experiment-I is described with particular emphasis placed on the neutronic characteristics, the control and instrumentation systems and plant safety considerations. The plan of operation is discussed and significant operating procedures are included. The site is described with reference to geology, climate, and population density.

Hazards which could result from both normal and abnormal operation are analyzed in detail with emphasis given to possible excursions and to the distribution of any resultant radioactive products.

It is concluded that a condition which would be hazardous to the surrounding area could result only from gross physical changes in the reactor core. It is further developed that such a change could result only from the concurrent failure of the control system and the catastrophic failure of the pressure vessel. These failures could be promoted by extremely unlikely occurrences such as sabotage, bombing of the facility, or the dropping of a large object into the pool. No individual operating error or equipment malfunction can result in the maximum credible accident (as defined in this report).

I. SUMMARY

A. GENERAL

The Gas-Cooled Reactor Experiment-I will examine the feasibility and characteristics of a compact heterogeneous, water-moderated, gas-cooled reactor for mobile power applications.

The reactor core consists of concentric tubular fuel elements which are composed of stainless steel and UO_2 "meat" clad with stainless steel. The concentric fuel tubes are contained in insulated aluminum pressure tubes through which the nitrogen coolant flows. The reactor assembly (73 pressure tubes and 2 plenum chambers) is immersed in a pool of water. The water acts as the moderator, the primary radiation shield, and the external core coolant. A lead fast-neutron reflector and gamma shield surrounds the core assembly. In normal operation at 2.2 megawatts the nitrogen coolant is heated from 800 to 1200°F as it passes through the core. Under these conditions the maximum fuel element temperature (including "hot spot" factors) is 1650°F.

In normal operation, the only radioactive waste product released from the system is coolant gas which may leak from imperfect closures and seals in the coolant loop. No health hazard is anticipated from this source.

*Written by the Military Projects Division of Aerojet-General Nucleonics, a subsidiary of Aerojet General Corporation, San Ramon, California.

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Operating procedures have been developed which insure a high level of personnel health and safety.

The prompt temperature coefficient of reactivity is estimated to vary from $-1.5(\pm1)\times10^{-6}$ $\Delta K/K$ per ^{o}C at ambient conditions, to $-0.9(\pm1)\times10^{-6}$ $\Delta K/K$ per ^{o}C at operating conditions. Flooding the core with water increases the reactivity by 10.8%. An excursion will not be inherently self-limiting short of fuel element melting or vaporization. The possibility of an excursion is minimized by restricting reactivity additions by control rod movement, or by deliberate core flooding to an addition rate less than 1.5% per minute.

A maximum credible accident is one that is caused by the simultaneous rupture of an underwater gas passage and the failure of the scram system with the reactor in a cold, near critical condition. If it is assumed that the core is flooded in 0.5 sec. then it can be shown that the resulting excursion energy is 380 megajoules, and the chemical energy is 100 megajoules. This accident will cause about 25% of the fuel elements to be vaporized and most of the remainder to be melted. The resulting explosion would be equivalent in force to the detonation of 38 lb of TNT and would destroy the core structure. Most of the fission products released from such an accident would condense, or otherwise be contained in the reactor pool. The reactor building would be virtually undamaged.

The wind and population distribution at the proposed location is such that there is a 90% probability that, in such an accident, the release of fission product activity into the atmosphere would not endanger populated areas. In the remaining 10% probability, the wind would carry the activity toward other NRTS sites or toward Atomic City (6 miles). The resulting radiation in these areas could be serious enough to warrant evacuation.

The accident described above, while credible, could result only from a highly improbable circumstance such as a bomb detonating in the pool near the reactor. The rupture of a major underwater gas passage would result in an excursion which would be limited by the control system to far less severe consequences than those described above. Similarly, a failure of the entire control system would result in a melt-down but not in an explosion. The simultaneous occurrence of these two events, coupled with a specific (cold, near-critical) condition of the reactor, is then credible but a highly improbable circumstance.

Other possible acccidents, such as a fuel element rupture or a coolant leak, might necessitate the evacuation and decontamination of the facility but such an occurrence should not result in any hazard outside the GCRE-I site.

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B. TABULATION OF FACILITY CHARACTERISTICS

1. Site

a.	Location	NRTS, Idaho
b.	Exclusion distance	75 ft (min) from reactor center line
c.	Prevailing wind direction	SW (day); NE (night)
d.	Population direction	SE (major); NW (secondary)

2. Operating Data

a.	Power (thermal, maximum)	2.2 Mw
b.	Power density	300 Kw/ft^3
c.	Core inlet pressure (nominal)	210 psig
d.	Core outlet pressure (nominal)	197 psig
e.	Core inlet temperature (nominal)	800°F
f.	Core outlet temperature (nominal)	1200°F
g.	Maximum fuel temperature (including hot spots)	1650°F
h.	Core cycle time (initial)	3000 hr
	(ultimate)	10,000 hr

3. Core

a.	Approximate dimensions (active section)	24" dia x 28" long
b.	Number of fuel elements (reference operating load)	61
c.	Number of control assemblies (all types)	14
d.	Hydrogen to Uranium (235) ratio	115
e.	Heat capacity in fuel elements	15000 cal/g per °C
f.	Volume fraction (in active core exclusive of reflector)	
	UO_2	1%
	Stainless Steel	10%
	Aluminum	6%
	Zirconium	3%
	H_2O Moderator	51%
	Insulation	9%
	Void	20%
g.	Uncooled fuel temperature Rate of rise at full power (max)	120°F/sec
h.	Reflector - Top	2" H_2O & 3½" Al
	Bottom	3½" Al
	Sides	4" Pb; H_2O.

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4. Nuclear Data

a.	Neutron flux - thermal (average)	3×10^{12} n/cm^2-sec
	fast (average)	1.9×10^{13} n/cm^2-sec
	max to avg ratio (thermal)	2.4 to 1
b.	Critical mass (cold, clean)	13.8 kg
c.	Total mass of fuel in new core (reference)	19.2 kg
d.	Neutron lifetime	50 msec
e.	Temperature coefficient of reactivity	
	Prompt at 60°C	$-0.9(\pm 1) \times 10^{-6}$ ΔK/K per °C
	Prompt at 20°C	$-1.5(\pm 1) \times 10^{-6}$ ΔK/K per °C
f.	Void coefficient	3×10^{-4}/cm^3
g.	Excess reactivity	
	Burnable poison	1.6% ΔK/K
	Shim control	2.0% ΔK/K
	Regulating control	0.4% ΔK/K
	Total	4.0% ΔK/K
h.	Reactivity worth of single fuel element (at core periphery)	0.40% ΔK/K
i.	Thermal utilization, f	0.655
j.	Resonance escape probability, p	1.0
k.	Thermal diffusion length, l	1.6 cm
l.	Fermi Age τ_{core}	92 cm^2
m.	Buckling, B^2	.0098 cm^{-2}
n.	Infinite multiplication factor k_∞	1.611

5. Fuel Assemblies

a.	Composition of meat	30 wt % enriched UO_2 in Type 316 stainless steel
b.	Cladding	0.006" Type 318 stainless steel
c.	Weight U-235 per element	315±5 gm
d.	Fuel assembly dimensions (overall)	1.85" dia. x 36" long
e.	Number of plates	12, formed into 4 concentric cylinders
f.	Coolant passages (cold)	.044 in. to .097 in.

6. Control Rods

a.	a.	Absorber	
		Safety	W, Cd
		Shutdown rods	W, In, Cd
		Setback, shim and fine rods	W
	b.	Blade Dimensions	
		Safety rods	7.75 x 21½ in.
		Shutdown rods	14.75 x 21½ in.
		Setback and fine rods	7 x 24½ in.
		Shim rods	10.25 x 24½ in.

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6. Control Rods - Continued

c. Reactivity worth, % ΔK/K

		Each	Total
Safety rods	(2)	1.5	3.0
Shutdown rods	(6)	2.5	15.0
Setback rods	(1)	0.3	0.3
Shim rods	(2)	1.0	2.0
Fine rod	(1)	0.4	0.4
Total			20.7

d. Maximum possible reactivity addition rates, % ΔK/K

	Per Min.	Per Inch
Safety rod	0.21	0.21
Shutdown rod	0.35	0.35
Setback rod	0.17	0.03
Shim rod	0.15	0.09
Fine rod	0.40	0.04
Gang I (2 Safety, 2 Shutdown)	1.12	1.12
Gang II (4 Shutdown)	1.40	1.40
Gang III (2 Shim)	0.30	0.18

e. Scram times

	Safety Rods	Shutdown Rods
Electronics	8 msec	8 msec
Break-away	~ 50 msec	~ 50 msec
Insertion	~ 200 msec	~ 750 msec
Total	~ 250 msec	~ 800 msec

7. Pressure Tubes and Plenums

	Tubes	Plenums
a. Dimensions		
Height	37 in.	~ 16 in.
Diameter (outside)	1.875 in.	28 in.
Wall thickness	0.058 in.	0.375 in.
b. Design Specifications		
Pressure (coolant)	200 psig	200 psig
Temperature (coolant)	1200°F	1200°F
c. Material	Al	304 stainless steel

8. Primary Coolant System

a. Coolant	N_2 with 0.5% O_2
b. Number of loops	1
c. Number of passes through core	1
d. Total volume of system	~ 225 cu ft
e. Heat transfer area of core	214.1 ft^2

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8. Primary Coolant System - Continued

f.	Heat flux	
	Average	33,160 Btu/ft^2
g.	Coolant mass flow rate (nominal)	63,000 lb/hr
h.	Maximum velocity in coolant passages	190 ft/sec
i.	Pressure	
	Maximum (compressor outlet)	242 psia @ 195°F
	Minimum (compressor inlet)	204 psia @ 140°F
j.	Temperature	
	Maximum (reactor outlet)	1200°F
	Mean	500°F
	Minimum (exchanger outlet)	140°F
k.	Maximum fuel surface temperature	1650°F
l.	Heat exchanger	
	Type	Shell & tube, single pass, gas-to-water
	Temperature (nominal)	Gas / Water
	Inlet	1200°F / 138°F
	Outlet	140°F / 150°F
	Heat Transfer	18 x 10^6 Btu/hr
m.	Compressor, Design Conditions	
	Suction	1220 cfm. @ 140°F & 198 psia
	Discharge	195°F and 242 psia

9. Secondary Coolant System

a.	Coolant	
	Type	Demineralized H_2O
	Flow rate	1000 gpm
b.	Heat Exchanger	
	Type	Shell & tube, single pass, water-to-water
	Pressure	
	Shell side	65 psig
	Tube side	150 psig
	Temperature, shell side (cooling water)	
	Inlet	75°F
	Outlet	105°F
	Temperature, tube side (demineralized water)	
	Inlet	138°F
	Outlet	100°F
	Heat Transfer	18 x 10^6 Btu/hr
c.	Pumps	
	Number	2
	Capacity (each)	1000 gpm
	Temperature at suction	100°F

10. Heat Sink

a.	Coolant	
	Type	H_2O
	Flow rate	2100 gpm
b.	Heat exchanger	
	Type	Induced draft cooling tower
	Temperature	
	Inlet	105°F
	Outlet	75°F
c.	Pumps	
	Number	2
	Capacity (each)	2100 gpm
	Temperature at suction	75°F

11. Summary of Accidents Investigated

a. Startup
b. Flooding
c. Fuel addition
d. External physical damage
e. Dropping material into the pool
f. Sabotage
g. Fluid temperature changes
h. Loss of coolant flow

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II. SITE DESCRIPTION

A. LOCATION

The experimental reactor is located in the Army Reactor Experimental Area, in the southern part of the National Reactor Testing Station, on the Snake River plain of southeastern Idaho. The site, as shown in Figure 1 is about 5 miles east of the NRTS Central Facility, 2 miles north of U. S. Highway 20, and 40 miles west of Idaho Falls. There is no agricultural activity within the NRTS area and virtually none within 20 miles of the site, as shown in Figure 2.

The residents within 20 miles of this site number less than 1000, most of whom live in Atomic City, 6 miles SSE. The frequency of wind toward this direction is about 2%. The other facilities at NRTS lie generally north and west of the site. The frequency of wind into this quadrant is about 7%.

Between 20 and 30 miles from the site, the total population is about 4000, mainly in the following localities.

	Distance (Miles)	Direction	Population
Howe	20	NW	50
Bernice	20	NW	25
Arco	25	W	1000
Moore	29	NW	300
Moreland	30	SE	400
Terreton	29	NE	300

Within fifty (50) miles, the population is about 110,000, of which 85% is located in the Snake River Valley south and east of the site at a distance of 30 to 50 miles. The wind frequency into this area is about 15%. Places with a population of over 500 within a 50-mile radius are:

	Distance (Miles)	Direction	Population
Atomic City	6	SE	500
Arco	25	W	1000
Blackfoot	33	SE	5200
Shelley	39	E	1900
Aberdeen	39	S	1500
Idaho Falls and Vicinity	40	E	30000
Mackay	45	NW	800
Rigby	47	E	1800
Pocatello and Vicinity	49	SE	31000

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The population within a 100-mile radius is about 210,000 of which 80% is along the Snake River.

B. GEOLOGY AND HYDROLOGY

The surface of the NRTS area is generally flat, except that near the southeast corner and at the northeastern edge there are a few cinder buttes which rise 200 or 300 ft. The elevation of the plain is about 5000 ft on the south and gently slopes to 4800 ft at the north. Streams flowing into the area are Big Lost River, Little Lost River, Birch Creek, and Medicine Lodge Creek. These go through shallow, meandering channels and disappear into the ground. No lakes or swamps have formed at these points.

Underlying the entire area is a series of Pleistocene lava flows, which are covered by soil or alluvium that varies in thickness from a few inches to more than 100 ft. At the GCRE site, the thickness varies from a few inches to 30 ft. Logs, taken of drilled wells, indicate that the successive lava flows occurred after considerable time intervals. The flows, which range in thickness from about 35 to 150 ft are frequently separated by layers of sand, gravel, or clay. The lava is uniform in appearance and character and apparently issued from fissures rather than from volcanoes. The lava was probably very fluid when hot, which could account for the large, nearly flat and level surfaces found in this area. The lava is porous and the flows frequently contain tunnels and cavities as the result of the top having cooled and formed a stationary crust while the hot interior portion continued to flow.

The surface drains toward a low area in which Big Lost River, Little Lost River, and Birch Creek formerly disappeared. Construction at the headwaters of these streams and irrigation diversions have moved the disappearance points upstream. Although there is a very moderate slope over most of the area, the site has good drainage for the limited rainfall. With the exception of Mud Lake, the area is well drained and devoid of swampy areas. The surface drainage at the GCRE site is southwest, with a 1% slope. This general slope is continuous; it drops approximately 80 feet in a mile and thus eliminates the possibility of water accumulation.

There is a large underflow beneath much of the NRTS area. The rate of flow is estimated to be at an average velocity of 1/2 mile per year. It is believed that the nearest point of surface discharge is along the banks of the Snake River below the American Falls Reservoir, 60 miles from the site, where numerous springs issue from porous lava beds.

Although the lava plain of the Snake River is geologically young, the surrounding mountains are mostly of great age. Some geologically recent faults appear to cross the lava-bed plains, although the traces are not evident on the surface. None of these faults indicate historically recent movement outside of the lava plain. Many recorded earthquakes have been felt in Idaho, but none were of sufficient intensity to cause more than minor damage to buildings. The seismic map of the United States shows that most of Idaho, including

the site is in zone 2; a zone where earthquake intensity sufficient to cause moderate damage may be expected.

It is believed that no serious damage is likely to occur to structures that are soundly designed to resist shocks of moderate intensity. Earth movements may break underground pipe connections at the faces of buildings, since this damage frequently occurs in moderate earthquakes.

C. CLIMATOLOGY

The Local Climatological Data is summarized in Appendix D. These data were taken at the Central Facility, 5 miles west of the proposed site. There are no local topographical features which would cause significant differences between the proposed site and the Central Facility. The climatological points which are most important for a hazards evaluation are:

1. There is usually an up-valley flow (southwest wind) during the day and a down-valley flow (northeast wind) at night. These flows occur about 80% of the time. This is fortunate because the population centers which might be affected by an accident lie almost entirely in a northwest direction (NRTS, Arco) or a southeast direction (Atomic City, Idaho Falls, Blackfoot, Pocatello). The nearest population center in the direction of the prevailing winds is the village of Terreton, 29 miles distant.

2. The normal annual precipitation in the area is less than 6 in., mostly in the form of scattered showers. Thus, the probability of cloud wash-out in the event of an accident is very low.

3. Temperature inversions throughout the lower 250 ft of the atmosphere are present for about half of the total time, and for a few hours almost every day. This condition would tend to hold any radioactive cloud to a low altitude, and retard cloud diffusion.

4. Destructive winds have not been a problem, but several thunder storms can be expected during each summer month, and lightning protection is required.

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III. THE REACTOR FACILITY (See Figures 3 and 4)

A. REACTOR DESCRIPTION

1. General Description

The reactor is composed of 73 fuel-containing pressure tubes which are rolled and welded into tube sheets which, in turn, are connected to the inlet (upper) and outlet (lower) plenum chambers. The reactor assembly, together with the control rods and neutron instrumentation assemblies, are bolted to a truss-type stand and submerged in a pool of water. This pool is thirty (30) ft deep and contains approximately 162,000 gallons of water which cannot be drained. Instruments are provided to control the level and to sound an alarm when the level is either too high or too low. The pool temperature is monitored and activity is continuously checked once per day. The pool water acts as the moderator, the primary radiation shield, and the external coolant for the pressure tubes. The pool water circulates by natural convection to control the temperature of the pressure tubes.

All underwater portions of the reactor assembly are designed for 200 psi and 1200°F coolant operation in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII. The components were fabricated under specifications which required the manufacturer to certify the material, qualify the welders, radiograph the welds, and proof test each component in accordance with the same code. Qualified inspection in the field assured compliance with these specifications.

Figure 5 shows the layout of the reactor, control rods, and support stand in the reactor pool. Figure 6 shows the detailed reactor design.

Results of the critical assembly tests indicate that a core of from 53 to 55 fuel elements will satisfy the operating requirements. However, to satisfy heat transfer and pressure drop requirements, a larger core is required. Design changes will be made in the fuel elements to permit the use of a 61 element core for operation at reference conditions. The 12 extra pressure tubes will be blocked with dummy graphite elements to avoid an inadvertently incorrect core loading. The loadings may be increased for experimental purposes but only after some modifications have been made to the control system and after the appropriate Technical Standards and Standard Operating Procedures have been revised and approved by the Supervising Representative for AGC and the Contracting Officer for the AEC.

2. Pressure Tube Assembly

The reactor pressure tube assembly is comprised of a bundle of 73 individual pressure tubes positioned by a 3.5 in.-thick aluminum tube sheet at each end. Each tube is rolled into the sheet and is welded at each end to provide a gas-tight seal. Sixteen, 1-in.-dia. removable aluminum stay rods are inserted in the tube sheet to prevent the pressure tube from

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being over-stressed.

The pressure tubes are made of 0.058 in. wall, 1.875 in. OD, seamless aluminum tubing, and are arranged in rows with 2.375 in. between row center lines and between tube center lines. Tubes in adjacent rows are displaced to form a triangular lattice as shown in Figure 7. The fuel element hold-down mechanism provides a means of orienting and locking each fuel element in place. This mechanism prevents the fuel elements from being ejected from the tube bundle if the inlet plenum or duct is suddenly depressurized.

A ring manifold, placed in the pool and directly encircling the top of the pressure tube assembly, is connected to the suction side of the pool water circulation pump. This collects the upward flow of pool water as it emerges from the pressure tube assembly. The fast-neutron reflector assembly acts as an outer wall for this "water jacket". The cooling obtained in this fashion maintains the temperature of the exterior surfaces of the pressure tubes at approximately 150°F.

The fuel elements are contained in the 37-in. long pressure tubes. The active part of the element extends 28 in. upward from the bottom tube sheet. Each fuel element is insulated to provide a heat barrier between the active fuel plates and the aluminum pressure tube. In addition, the tube sheets are insulated from the hot coolant gas with 3/4 in. thick Refrasil cloth protected by a 0.005 in. stainless steel scuff sheet.

The region of maximum stress in the tube bundle assembly appears in the center pressure tube. The pressure tube walls are subjected to the following stresses:

	Calculated	Allowable
Tensile stress	-3500 psi	-5800 psi*
Hoop stress	3230 psi	5800 psi*
Shear stress	3350 psi	3800 psi

*The allowable values refer to ASME Code. As such, a safety factor of approximately 5 is inherent in the stress values.

3. Plenums and Connections

The inlet and outlet plenums are type 304 stainless steel cylindrical chambers with a wall thickness of 0.375 in. These are joined to the pressure tube assembly by bolted flanges which are sealed with Type 3003 aluminum gaskets. The inlet plenum is fitted with a flanged cover which may be remotely removed to allow access to the core for fuel handling and maintenance operations. Both plenums are lined with nominally 0.5-in. thick laminated stainless steel radiation barrier type insulation, composed of a series of 0.016-in.-thick stainless steel sheets. Dimples formed into the sheets maintain the spacing essential for good insulation and provide sufficient gas passages to prevent collapse of the assembly when the system is depressurized.

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Both plenums are fitted with 10-in. IPS nozzles to join with the inlet and outlet coolant gas ducts. The ducts are flanged and bolted to the nozzles with nickel plated, Type 304, stainless steel ring gaskets.

The ducts are 10 in. stainless steel, schedule 80, pipe that is insulated with 1 in. of stainless steel, radiation barrier type, insulation similar to that used in the plenums. This insulation extends from the plenum to a point beyond the exterior surface of the reactor pool wall. Outside the pool the ducts are externally insulated with standard magnesium oxide insulation. The internal and external insulation are overlapped for 8 ft. to minimize thermal stresses. An expansion bellows seals the inlet duct to its supporting sleeve to relieve the differential thermal expansion in the inlet and outlet ducts. The outlet plenum is also fitted with a 6-in. nozzle that is concentric with a 3-in. nozzle. This provides a reactor flooding line in the annulus and the drain and steam injection line in the central nozzle. The steam line is baffled to prevent live steam from impinging on the fuel elements and the assembly is designed for complete draining.

The inlet plenum is fitted with four 3-in. nozzles through which the thermocouple leads from the instrumented fuel element leave the pressure vessel. An instrument flange (shown in Figure 6) is inserted between the lower tube sheet and the exhaust plenum. This flange is fitted to accommodate the thermocouple leads and gas sampling tubes that sense coolant conditions from each fuel element.

4. Reactor Reflector and Gamma Shield

Surrounding the reactor core is a nominally 4-in.-thick fast-neutron reflector and gamma shield made of lead. This reflector is made up of several sections of lead that are cast around 1/2-in. diameter aluminum tubing. Gamma heat generated in the lead is removed by a flow of 25 gpm of water in these tubes. Each lead section has protective upper and lower scuff plates and is spray-coated with Metco and aluminum paint for protection against galvanic corrosion. The reflector assembly contains slots to allow passage of the control rod blades and guides. The lead sections in the vicinity of the control rods are supported by the control rod guide assembly. The remainder of the lead is supported by an aluminum stand that is attached to the lower tube sheet. This reflector-shield assembly is positioned 1/2 in. from the outermost pressure tube and is designed so that it can be removed and replaced remotely. A 3-in. space above and below the lead is provided to allow convection of the cooling moderator.

5. Reactor Stand

An aluminum truss, as shown in Figure 5, supports the reactor in the reactor pool. This truss is hung from clamps that are set into the pool wall concrete. This design provides the flexibility that is required to accommodate the thermal expansion of the gas ducts.

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The control rods are mounted on the same truss to assure the alignment between the control rods and the reactor. In addition the truss serves to support the reactor neutron instrumentation assemblies and certain other reactor auxiliaries.

B. FUEL ELEMENTS

1. Description

The reference fuel element for GCRE is shown in Figure 8. Neutronically, this fuel element consists of 315±5 grams of U-235 (as [redacted] enriched UO_2) dispersed in Type 316 stainless steel to form a 33-mil-thick "meat" section of 30 weight percent UO_2. The "meat" is clad with 6 mils of Type 318 stainless steel. The active length is 28 in. and the active material is distributed through twelve fuel plates which are assembled into four concentric cylindrical "tubes" in each fuel element.

Mechanically, the fuel element is assembled about a central spine. A bottom fitting is screwed and pinned to the spine. Near the top, the fuel plate support spider is screwed onto the spine and pinned. The fuel plates are slotted to accommodate the support spider arms. These slots are designed so that a pin, inserted after all fuel plates are hung on the spider, locks the plates to the spider arm. The plates are restrained in guide grooves in the bottom fitting which are deep enough to permit downward expansion.

Each of the four concentric fuel tubes is composed of three curved segments. The ends of the segments are restrained by stainless steel clips which hold the plates in the proper orientation to form a cylindrical tube and which maintain the gaps between tubes. The joints in the various tubes are staggered circumferentially.

The fuel plate and spine assembly is enclosed in a stainless steel inner liner which is supported from the bottom fitting and restrains the fuel plate locking pins described above. This liner is free to expand upward into a guide groove in the top fitting. The outer liner consists of three cylindrical sections: a) the upper section of stainless steel welded to the top fitting; b) the lower section of stainless steel screwed and pinned to the bottom fitting; and c) the central section of zirconium supported by the upper and lower sections and free to slide on the inside surface of both. The burnable samarium poison in the form of a sheet 5 mils thick is riveted to the central section. The space between the inner and outer liners is filled with Thermoflex insulation. The inner liner is perforated near its bottom end to permit water to drain out of the insulation after a flooding operation.

Coolant flow distribution across the core is controlled by a snap-fit orifice in the bottom fitting of each fuel element. This orifice may be removed and installed in active elements by the use of a special underwater tool.

2. Testing Program and Results

The reference fuel element was designed with the aid of the results of various developmental tests. These tests included, capsule tests of the fuel plate material, corrosion studies of the cladding materials, heat transfer and fluid flow measurements, and mechanical tests of the fuel element components. The final design of the element was tested by thermally cycling a model of the element, and by two in-pile tests that were designated I-1T and I-1'T. Another in-pile test, designated I-3P, is scheduled for irradiation beginning in March 1959. These developmental tests are briefly described, and the in-pile tests to study the gross characteristics of the element are presented in somewhat greater detail in the following sections.

a. Capsule Test

Several capsule tests on the UO_2-stainless steel specimens of the type used in the GCRE fuel element were performed by Battelle Memorial Institute. One of the tests was conducted at 1650°F and contained four specimens. A detailed examination of these specimens showed a 4.5 atom % U235 burnup and that all of the specimens were in good condition. Another capsule test that contained two 30 wt % UO_2 specimens showed a 7.2 atom % burnup at 1450°F. Although these specimens were blistered, neither had failed. These capsule tests indicate that the fuel plate materials can attain a life of about 4000 hr in GCRE-I when operated at about 1650°F. This corresponds to the maximum calculated hot spot in the element.

b. Corrosion Tests

Corrosion tests were made on the AISI 318 stainless steel alloy (which is used for cladding the GCRE fuel plates) both by BMI and by Aerojet. In a test of about 2600 hr duration in the reference gas, (nitrogen plus 0.5 volume% oxygen) at 1650°F the corrosion depth was 0.8 mil. The test was conducted at one atmosphere pressure. In a test made at a pressure of 200 psi, the corrosion rate was accelerated. The results of these corrosion tests appear to indicate that the 0.006-in.-thick cladding, operating at 1650°F in the GCRE atmosphere, can last for about 8000 hr.

c. Mechanical Test

Mechanical tests were made of the various components of the fuel element. The tests made of mechanical supports for the fuel element indicate that they are capable of withstanding a pull in excess of 1000 lb at room temperature.

Pressure tests were made to verify experimentally that neither the Thermoflex insulation nor the insulation liner would be damaged if the loop were suddenly depressured. In these tests, the fuel element was placed in a 225 psig nitrogen environment at room temperature and explosively depressured. No damage was noted.

The dimple on the inner insulation liner that is used for retaining the gap spacing between the A plate and the inner insulation liner was tested for strength. The tests were made in a furnace at 1650°F by loading each dimple to about 3 lb. The dimple heights measured both before and after the test indicated a decrease in height of the hemispherical dimple of 0 to 1 mils. This is a about the load that is anticipated in the lower temperature operating condition and thus it is believed that the strength of the dimples is adequate.

The machined clip sections were tested with a load of 448 psi at 1650° and showed no decrease in height.

d. Thermal Cycling Test

A model of the element that contained 25 wt % natural UO_2 fuel plates was thermally cycled 16 times in a vertical position. During the heating cycle the assembly was placed in a furnace in an argon atmosphere and the top of the element was heated to 1550°F and the bottom to 1350°F. After a temperature equilibrium was established, the assembly was removed from the furnace and allowed to cool in a vertical position at room temperature. After 16 of these cycles, the dimensions of the gap remained within tolerance. The model appeared to be mechanically satisfactory.

e. Flow and Heat Transfer Tests

Heat transfer and fluid flow experiments were conducted to supplement the existing data on the flow in annular channels. The test section used in the investigation consisted of two stainless steel tubes, 28 in. long, arranged concentrically to form an annular flow passage. The inner tube had a 1.018 in. OD and the annular flow passage was 0.071 in. wide. Both the inner and outer walls of the annulus were heated electrically. The heat flux was constant over the entire length of either wall during the test. However, the heat flux at each wall was varied from test to test to simulate GCRE operating conditions.

Experimental studies were conducted to establish the characteristics of the heating of symmetrical and asymmetrical concentric tubes. The experimental data was found to follow theoretical predictions in areas where literature was available. Where no theory could be found to permit analysis of the situation (e.g., where severe asymmetry permitted one tube to touch the other), heat fluxes and temperature rises were found to be quite moderate.

f. I-1T In-Pile Test

An in-pile experiment was conducted on a fuel element, designated as I-1T. This element was placed in the BRR-GCR loop which is located adjacent to the Battelle Research Reactor core, to determine the effects of reactor flux on an operating element. The purpose of this test was to gather engineering data. The test was limited to 1180 hr and operated

at an average power output of 13 kw. The flow rate was maintained at approximately 250 lb/hr to attain a fuel plate temperature of 1400°F. Because of the neutron flux limitations in the BRR, the I-1T element could not be tested at 30 kw which more closely simulates GCRE conditions. However, the I-1T element was operated for 90% of the time at higher temperature (1400°F) in a radiation field with encouraging results.

Specific results of the in-pile experiment which are pertinent to this report are: (1) heat transfer calculation methods are accurate for predicting element surface temperatures and operating characteristics; the deviations from the predictions were less than the instrument sensitivity; (2) no geometrical changes were detected, although the element was thermally cycled about 35 times; (3) I-1T flux depression measurements are in good agreement with those of GCRE critical experiment; (4) heat transfer coefficients that were calculated from the derived data averaged 12.5% higher than those calculated from the Colburn equation. (This agrees with the heat transfer coefficients that were measured in the heat transfer experiment described above); and (5) transition flow was not detected down to a Reynold's number of 4000.

g. I-1'T In-Pile Test

An in-pile test was made on a second in-pile element designated I-1'T primarily to study the effects of thermal cycling. This test was a sequel to the I-1T test and used an idential test element placed in the same position in the BRR-GCR. The element was subjected to 502 thermal shocks in a period of 393 hr. A total of 453 thermal cycles was accomplished by varying the flow rate between 200 and 1000 lb/hr with corresponding temperature changes between 1600 and 700°F. These cycles were not continuous but were done in spurts over short periods of time and with a limited number of cycles per period. The number of cycles per period increased as the in-pile test neared the end, until as many as 53 continuous cycles were made over a 150-minute period on the day before termination of the test. The rate of temperature change during the thermal cycling varied from 7 to 10.5°F/sec.

Forty-nine thermal shocks were due to scrams, power reductions and normal shutdowns of the reactor. The bulk of the 453 flow controlled cycles occurred in the latter period when the cycling periods were longer and more severe. Thermal cycling was discontinued after 370 hr.

Hot cell examination of these elements revealed no external evidence of mechanical damage to the fuel plates. Two of the fuel clips were badly corroded in the vicinity of the hot spot but this is of no concern as the clips were fabricated from AISI Type 302 material instead of the reference AISI Type 318. The fuel plates were neither distorted nor significantly corroded and none of the fuel plates had escaped from the clips. During the test, coolant activity data indicated no ruptures of cladding or significant fission product diffusion.

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The irradiation was carried out under flow condition to give a nominal peak temperature of 1600°F (as compared to the GCRE-1 nominal peak temperature of 1350°F). Under this condition a hot spot of 230°F was recorded on the "C" plate. The cause of this hot spot has not as yet been determined but its magnitude is within the provided hot spot allowance (300°F).

h. I-3P In-Pile Test

The I-3P in-pile test will be a 4000 hour test of a GCRE-I design, production element. It is planned to be conducted in the BMI-16 loop at the ETR beginning in March 1959. This will be a relatively long test (eight months) and its principle purpose is to determine the approximate element lifetime characteristics and assess the mechanism of failure. With the crude lifetime predicted from the operation of the I-3P element, the performance of the element in the GCRE-I can be estimated.

The operating conditions of the I-3P test will be referenced to the GCRE core fuel element with respect to peak power production. The ETR loop will be shimmed with poison to a point where power production in the fuel element is approximately 45 kw. average (maximum contemplated element power in GCRE at 2 Mw). During the ETR cycle, due to the core burnup, the flux level is expected to change by a factor of 1.5 and the power will vary accordingly. The peak power will be about 56 kw.

3. Storage

Fuel elements for GCRE-I will be stored only in one of three places:

1. The new fuel storage vault. Here, unirradiated fuel elements will be stored in their shipping containers under the protection of an ADT system. No material other than fuel will be stored in this limited access vault. The fuel element shipping containers have been designed to preclude any problem of critical array in the dry or wet condition. Elements will be transported from the vault to the reactor in a special cart which is designed to prevent the attainment of a critical array, even in a flooded condition.

2. The fuel element storage rack. This consists of a series of tubes that are supported in a rigid framework mounted in the reactor pool. Irradiated fuel elements, removed from the reactor, will be stored here until required for re-use or until shipped to a chemical processing facility. The design of the rack precludes the establishment of a critical array. Covers are provided for the tubes to prevent the possibility of security infractions.

3. The reactor. Fuel elements being tested or being used in the operation of the reactor will be identified by serial number and exact records will be maintained of the history and of the location of each element at any time.

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4. Fuel Element Temperatures

At design operating conditions the hottest point on the surface of the hottest fuel element plate is calculated to be 1350°F (neglecting hot spot factors). In GCRE-I two instrumented fuel elements will be used to measure the peak or near peak temperatures. One of these elements is located near the center of the core and the other near the periphery. Figure 9 shows the calculated profiles of the average heat generation per unit length, fuel plate temperatures, and coolant temperatures.

Hot spot analysis indicates that some local areas could operate in temperatures as high as 1650°F. Hot spot factors of 1.55 on the coolant temperature rise and 1.53 on the film temperature rise give this maximum possible temperature. Hot spot factors are used to estimate the highest temperatures that can occur due to uncertainties in flow conditions, heat generation, fuel element fabrication, and other uncertainties due to empirical data or manufacturing tolerances. Hot spot factors are used in a manner that assumes the simultaneous occurrence of worst-case events on the hottest spot of the element. Therefore, the maximum temperatures determined with these factors are conservative and are not likely to exist. The table which follows tabulates the individual hot spot factors that were used to arrive at the major factors.

TABULATION OF HOT SPOT FACTORS

	$F_{\Delta\theta}$	$F_{\Delta t}$	$F_{\Delta T}$
Plenum flow distribution	1.04	1.03	--
Warping	1.04	1.04	--
Thickness	1.02	1.01	1.10
Fuel dispersion	--	1.05	1.05
Flow channel	1.16	1.04	--
Core flux distribution	1.15	1.15	1.15
Fuel element flux distribution	1.05	1.05	1.05
Thermal condition of fuel	--	--	1.04
Thermal condition of cladding	--	--	1.02
Convective heat transfer coefficient	--	1.07	--
Totals	1.55	1.53	1.48

$F_{\Delta\theta}$ = Bulk rise hot spot factor
$F_{\Delta t}$ = Film rise hot spot factor
$F_{\Delta T}$ = Plate rise hot spot factor

C. REACTOR CORE

The pressure vessel and fuel elements which were described earlier are combined to form the reactor core. The neutronic characteristics of the 61 element core are reviewed in the summary tabulation (Section I, B). The heat transfer and fluid flow characteristics are discussed below.

Steady state GCRE-I fuel element temperature profiles were calculated by the use of an IBM 650 computer. The IBM code that was used neglected axial conduction; assumed constant moderator temperature, but included thermal radiation, heat transfer to the moderator and heat flux asymmetry. This code has been checked against experimental data and has proved to be accurate. All heat transfer analyses were based on a 61 element core.

In optimizing the GCRE-I fuel plate spacing, a ground rule was adopted that all maximum plate temperatures be the same. This results in a lower maximum fuel element temperature, which is desirable because of the limitation upon lifetime imposed by fuel element temperature.

The fuel element temperature profiles, i.e., the four fuel plates, the four gas passages, and the stainless steel liner, are illustrated in Figure 9. Also shown in this figure is all heat transfer information pertaining to the fuel element such as gap widths, areas, equivalent diameters, heat transfer coefficients and flow rates. The power profiles shown are from GCRE-I critical assembly data with the burnable poison in the core. From this Figure it can be seen that the maximum nominal surface temperature in the element, under normal operating conditions, is about 1350°F. The temperature difference between plates at the point of maximum temperature is about 10°F maximum.

The water (moderator) is circulated through the external core and 25 gpm circulates through the fast-neutron reflector. The average temperature of the reflector will be approximately 300°F.

If the coolant pump should fail, localized boiling would occur on the inside surface of the reflector and on the pressure tube surfaces. This would cause the water to become less dense and thus cause the reactor to go subcritical and shut down.

The gas coolant passages in the GCRE-I fuel elements will be orificed so that the maximum nominal surface temperature of each element is the same. The orifice sizes range from 1.115 in. smooth bore in the center element to 0.550 in. on the reactor periphery. Corresponding element flow rates are 1387 lb/hr and 790 lb/hr. To obtain the necessary experimental data on element pressure drop, flow tests were run on an experimental element, (FDU-4). The results of these tests were extrapolated to the GCRE element. By using the power plots from the GCRE-I critical assembly, orifice sizes for all elements were determined The core pressure drop is 12.4 psi and with a 0.4 plenum pressure drop, the total reactor pressure drop is 12.8 psi.

The following power measurements for GCRE I were obtained from the GCRE-I critical assembly:

Radial - $\frac{q_{max}}{q_{avg}} = 1.30$ $\frac{q_{min}}{q_{avg}} = 0.69$

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Axial - $\frac{q_{max}}{q_{avg}} = 1.41$

Element - $\frac{q_{max}}{q_{total}} = 0.44$ $\frac{q_{min}}{q_{total}} = 0.12$

These power measurements were used in all heat transfer and pressure drop analyses.

D. CONTROL SYSTEM

1. Nuclear Characteristics

While the potential hazards from GCRE-I are, in general, similar to those of other reactors of the same power range, several special characteristics unique to GCRE-I should be described here because of their effect on the reactor control system.

a. The fuel elements are thermally insulated from the water moderator. In an excursion, the moderator will be heated only by a small amount from the fast-neutron and gamma heating effects. The prompt temperature coefficient of reactivity is $-0.9(\pm 1)\times10^{-6}$ ΔK/K per ^{o}C at operating conditions (60^{o}C) and $-1.5(\pm 1)\times10^{-6}$ ΔK/K per ^{o}C ambient conditions (20^{o}C). Thus, an excursion will not be self-limiting as a result of its temperature coefficient of reactivity but can continue until gross deformation of the reactor core occurs. For this reason an extremely reliable control system is required.

b. Because the reactor will be under-moderated during normal operation, fast flooding of the coolant tubes will result in a total reactivity increase of approximately 10.8% ΔK/K. Sufficient control is required to counteract this reactivity increase.

c. During operation, some portions of some of the fuel elements are expected to be near their maximum permissible temperature of 1650^{o}F. The reactor power must automatically be decreased at power levels or temperatures only slightly in excess of nominal. To avoid excessive outage time, a "setback" system is indicated. Such a system should be designed to operate very close to the normal operating conditions and be backed up by a scram system.

d. Since the reactor is an experimental device, the control system provided must be flexible enough to accommodate certain possible changes and additions in the reactor.

e. In order to reduce flux perturbations and to maintain a more nearly uniform temperature distribution, the thermal neutron absorption of control rods within the reactor core during operation must be held at a minimum.

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In consideration of the above requirements, the control system for GCRE-I includes the following components:

(1) Samarium which is dispersed in stainless steel foil and is mechanically attached to the interior surface of the zirconium section of the fuel element outer liner. This burnable poison, which is worth about 1.6% Δ K/K initially, will balance the loss of reactivity due to fuel burnup and to the accumulation of fission products other than xenon to within 0.3% over the expected core lifetime.

(2) Two safety rods each with a total insertion time of approximately 250 milliseconds. Each of these rods controls 1.5% Δ K/K.

(3) Six shutdown rods operating from the scram system in about 0.8 sec. These rods together control 15% ΔK/K.

(4) A setback rod to reduce power by 30% within 3 seconds and by 70% within 90 seconds. This rod is worth 0.3% ΔK/K.

(5) Two shim rods to balance xenon buildup and to correct for minor perturbations in reactivity that result from temperature changes and burnable poison depletion. Each of these rods controls 1.0% ΔK/K.

(6) A regulating (fine) rod, worth 0.4% ΔK/K which is capable of being manually operated or which can be controlled by a servo system.

The total worth of the above system, neglecting the samarium poison, is 20.7% ΔK/K in a cold, clean reactor. This is adequate to control the 2.4% loading excess and the 10.8% increase which accompanies flooding. The maximum rate of addition is physically limited to less than 1.5% Δ K/K per minute. If the period scram should fail, this rate of increase through criticality would place the reactor on a period of 50 milliseconds as the power level reached 2 Mw. This is an undesirable, but not hazardous, situation since the scram system will prevent reactor damage if the period at 2 Mw is as short as 20 msec. Procedural controls (see ANTS 102, App. E-1; ANSOP 5702, Appendix F; and Figure 10) restricts the addition rate to 0.1% ΔK/K per minute when the k_{eff} is greater than .995. This addition at criticality would result in a period of 5 seconds at 2 Mw.

The rod absorbers for the shim, setback and fine rods are chromium plated Mallory (90% tungsten, 5% each copper and nickel) alloy. The shutdown rod absorbers are built-up assemblies of a Mallory alloy core sheathed in cadmium-indium foil and enclosed in a stainless steel sandwich. The safety rod absorber is a built-up assembly of a Mallory alloy core encased in a stainless steel sandwich which has been metallized with cadmium on the interior surfaces.

Because the neutron flux is not uniform at all places in the core, the reactivity worth per unit distance of travel of the rods varies

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depending on the degree of insertion into the flux field. The data from the preliminary rod calibration tests conducted at BMI reveals, for example, that half the worth of the safety rods is not realized until 80% of the rod is in the core. This situation is recognized in the preparation of procedures relating to rod motion (start-up, etc.) and is shown graphically in Figure 11.

2. Interlock and Safety Circuits

Portions of the system operate automatically under the control of the setback, scram or servo circuits. In addition, interlocks are provided to assure that no rod is withdrawn unless the reactor is in operating condition; and, that once initiated, the rods are withdrawn in an established sequence. These safety and interlock circuits are described below:

a. Rod Withdrawal Permissive Interlock Circuit

This is a series circuit which consists of micro switches (or relay contacts) as indicated below. Individual switches or contacts may be bypassed as indicated and the whole chain is automatically by-passed when rod magnets are energized. Procedures covering the deactivation of these interlocks are shown in ANSOP 5711, Appendix F.

(1) All rods must be in the full-in position.

(2) A significant count rate must be indicated by the start-up count rate recorder. This up-scale recording closes a micro switch in the recorder. This count rate will be generated by a source which is capable of delivering 20 cps from the start-up count rate systems. The source is contained in a specially designed dummy fuel element and may be placed in any pressure tube in the core. The normal operating position will be I-[illegible]3 (see Figure 7). The source will be in the reactor at all times when fuel is present.

(3) The coolant gas temperature leaving the heater must be at a specified level, closing a microswitch in the temperature recorder. This interlock has a key-operated bypass switch.

(4) The reactor pool water level must be at a specified level, energizing a control relay with contacts in this interlock circuit. This interlock has a key-operated bypass switch.

(5) A secondary (demineralized water) coolant circulation pump must be in operation. A relay in the pump motor starter circuit closes a set of contacts in the interlock circuit. This interlock has a key-operated bypass switch.

(6) A heat sink (cooling water) coolant circulating pump must be in operation. A relay in the pump motor starter circuit closes contacts in this interlock circuit. This interlock has a key-operated bypass switch.

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(7) The gas coolant pressure must be at a specified level, closing a microswitch in the pressure recorder. This interlock has a key-operated bypass switch.

b. Scram Circuit

This is a series circuit which consists of microswitches and relay contacts as indicated below. All control components of this circuit must be energized in order to energize the safety and shutdown rod magnets and consequently, the de-energization of any single component de-energizes the magnets and scrams the rods. In addition, deactivation of this circuit results in the non-scrammable rods (shim, setback and fine) being driven into the reactor at maximum speed. Rules for deactivation of this circuit are presented in ANSOP 5711, Appendix F.

(1) The five run-safe switches in the reactor area must be in the run position. This series circuit energizes a relay with contacts in the scram circuit. Opening any switch scrams the rods.

(2) The gas coolant flow must be at a specified level, closing a microswitch in the flow recorder. This circuit has a key-operated bypass switch. Flow reduction to 85% of specified value scrams the rods.

(3) The reading on the Log-N recorder must be less than a specified value in order not to open the normally closed microswitch in the recorder. This circuit has a key-operated bypass switch. A reading 15% above the specified valve scrams the rods.

(4) The reading on the period recorder must be greater than a specified value in order not to open the normally closed microswitch in the recorder. This circuit has a key operated bypass switch. A period as short as 7 seconds scrams the rods.

(5) All doors to the reactor area must be closed. This series circuit energizes a relay with contacts in the scram circuit. This circuit has a key operated bypass switch. Opening any door scrams the rods.

(6) The seismic switch must be closed. Opening the switch scrams the rods.

(7) The nitrogen pressure to the scram cylinders on the safety rods must be above a specified level closing a set of contacts in the scram circuit. A pressure less than the specified value opens the contacts and scrams the rods.

(8) The gas coolant temperature leaving the reactor must not exceed a specified value in order not to open the normally closed microswitch in the temperature recorder. A temperature 60°F higher than the specified value scrams the rods.

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(9) The normal source of electrical power to the facility must be energized to close "power failure" contacts in the scram circuit.

In addition to the above, the composite safety amplifiers (see Section III, E) will de-energize the magnets and scram the rods if the power level is 15% above the specified value or if the period is as short as 7 seconds. A manual scram button is also provided to interrupt the circuit and scram the rods.

c. Setback Circuit

This is a parallel circuit consisting of micro-switches and relay contacts to energize a control relay when any one of the component circuits is energized. The control relay bypasses the normal control on the setback rod and energizes the high speed motor to drive the rod in.

(1) The gas coolant flow must be greater than 90% of the specified value. A flow as low as 90% of the specified value closes a microswitch in the flow recorder, actuating the setback circuit.

(2) The power level as recorded on the power level recorder associated with the safety amplifiers must not exceed 110% of the specified value. Exceeding this value closes a microswitch in the recorder and actuates the setback circuit.

(3) The period, as monitored by the startup neutron detector system (see Section III, E) must be greater than 10 seconds. A period shorter than 10 seconds actuates the setback circuit.

(4) The gas coolant temperature leaving the reactor must not be more than 40°F above the specified value. A higher temperature closes a microswitch in the temperature recorder and actuates the setback circuit.

In addition to the above, a manual setback button is provided on the console to actuate the setback circuit.

d. Rod Withdrawal Interlocks

In order that the rods are withdrawn in the proper sequence, interlocks are provided to prevent improper action during rod withdrawal. These interlocks are described below:

(1) The setback, shim or fine rods may not be withdrawn until all shutdown rods are full out.

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(2) The shim or fine rods may not be withdrawn until the setback rod is full out.

(3) The fine rod may not be moved until the shim rods have been withdrawn approximately 0.5 in.

(4) Four shutdown rods (other than those on Gang I above) may be moved as a gang (Gang II).

(5) Both shim rods may be moved as a gang (Gang III).

(6) A selector switch on the console prohibits the simultaneous motion of two rods except for simultaneous movement of the rods in gangs I, II and III described above and the simultaneous movement of shim and fine rods for interchange purposes.

e. Fine Rod Automatic Control

Under certain conditions, the fine rod may be placed in automatic operation. In this condition, the rod will move under the signal of a servo unit to alter the reactor power so that a preset signal is balanced by the output of the linear power system (See Section III, E). In order to place the fine rod on automatic operation, the following conditions must be satisfied:

(1) The scram circuit must be energized
(2) The setback circuit must be de-energized
(3) The fine rod must not be at either extreme of travel

3. Mechanical Description

As indicated above, the control system consists of twelve rods. These are divided into two control units, one of which is mounted on the reactor support stand on either side of the reactor.

The rods are divided between the two control units as follows:

Unit No. 1 (West Bank)
No. 1 Safety Rod
No. 1, 2 and 3 Shutdown Rods
No. 1 and 2 Shim Rods

Unit No. 2 (East Bank)
No. 2 Safety Rod
No. 4, 5 and 6 Shutdown Rods
Fine Rod
Setback Rod

Mechanical characteristics which are common to all the rods are as follows:

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a. All absorber blades enter the reactor core horizontally. The blades are mounted vertically and aligned so that they enter the core in the spaces between the pressure tubes.

b. The absorber blades of all rods are supported for their full stroke in top and bottom guide rails. The guide rails are Sanforized hardened aluminum in all cases except the safety rod guides, where Electrolized stainless steel is used. All of the blade guides are rigidly supported in an aluminum guide holder assembly.

c. In all cases the blades are connected to the actuator with a latch assembly. This latch is designed to accommodate reasonable amounts of angular and parallel misalignment and to withstand the high forces that are developed in acceleration and deceleration of the rod drives. The latch is equipped with a leaf spring lock which may be disengaged remotely.

d. The actuator mechanism is sealed from the pool water by a housing. The seal is provided by two rings of chevron packing in the case of the safety and setback rods, and by double "O" rings for all other rods. A drain connection between the two seals carries any leakage water to a drainage collection tank which serves all the rods in the control unit. An electrode type liquid detector in the drain tank sounds an alarm in the control room to notify the operator that water has accumulated in the tank. In this event, the water can be remotely blown back into the reactor pool. A separate electrode type liquid detector in each of the actuator mechanism housings sounds an alarm in the control room if water leaks past the second seal. The double "O" ring seal has been tested by continuous cycling for 1000 hours under water with no evidence of leakage.

e. The microswitches which indicate extremes of rod and actuator travel, energize and de-energize the drive motors, and function in the interlock circuits are all located in the sealed housing. These are cam operated switches, actuated either directly by the rods or by cams on a chain drive indicator lead screw.

f. The electrical wiring for the motor controls, magnets, and limit switches for each rod is contained in a flexible, armored waterproof sheath that is sealed to the actuator mechanism housing and suspended from the pool wall above the pool water level.

A description of the significant features of each type of rod actuator mechanism is presented below.

a. Safety Rod (See Fig. 12)

This rod is withdrawn by an electric motor operating through a lead screw and nut. The blade is connected to the rod extension by a spring loaded latch and this assembly is coupled to the drive by a solenoid-actuated ball-detent latch. During withdrawal, a spring is compressed by a piston which is integral with the rod extension and is contained in an open-ended cylinder. When the rod is "cocked", the spring is compressed and the

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entire housing is pressurized. The rod is held out only by the ball detent. When a scram signal occurs, the solenoid (magnet) is de-energized and the ball detent is retracted, which releases the rod under the combined driving force of the compressed gas and the spring. The rod is snubbed by pool water which enters the cylinder in front of the piston and is ejected through properly placed ports as the piston is driven in. The motor driven portion of the actuator is subsequently driven in in 15 minutes to retrieve the rod by means of the ball detent latch.

The pressure for rod actuation is drawn from a bank of 96 bottles of nitrogen gas. The pressure in the rack manifold is monitored by a pressure transducer which sounds an annunciator and initiates a scram upon low pressure. The supply lines are independent for each rod and each housing is fitted with check valves so that the driving pressure is retained for one scram even if the supply fails.

b. Shutdown Rod

This actuator mechanism is similar to that of the safety rod except that; (a) it is not pressurized, and (b) a nickel plated magnet is used in place of the ball detent.

c. Setback Rod

The actuator for the setback rod consists of two motors. To withdraw the rod a magnetic clutch is energized and a non-reversing motor drives a lead screw and nut assembly. As soon as the rod is full out, the magnetic clutch is de-energized. When the setback signal occurs, a second 1/2 horsepower 10,000 rpm motor drives the rod in less than three seconds. This unit has been cycled over 500 times with no failures.

d. Shim Rod

This rod is actuated by a two-phase reversing motor driving a lead screw and nut assembly.

e. Fine Rod

The actuator for this rod is similar to the shim rod except the gear ratio in the drive chain is different.

Figure 13 summarizes the significant neutronic and physical characteristics of the control rods.

E. INSTRUMENTATION (See Figure 14)

1. Nuclear Instruments (See Figure 15)

The nuclear instrumentation of GCRE-I consists of nine essentially independent channels complete with detectors, amplifiers, recorders and associated interlock circuitry. These nine channels span the entire range

of power from .001 watts to 10 megawatts. In some cases, these channels serve as back-ups, one for the other. The following description combines the back-up channels into systems for discussion.

a. Start-up System (Channels I and II)

Range - .001 to 50 watts
Detectors - Fission chambers (2)
Amplifiers - Linear pre-amplifiers and linear amplifiers and discriminators (2)
Data Presentation - Binary scaler and timer driven from either amplifier system through a selector switch
- Log count rate meter for each channel
- Log count rate recorder for either channel with selector switch
- Period recorder for either channel through selector switch (shared with Log N system)
- Log count rate meter for either channel with selector switch - slaved from recorder

Auxiliary Circuits - Period detection chassis for both channels
- Remote manually operated neutron shields for each fission chamber
- Automatic and manual fission chamber high voltage disconnect (see Log N system below)

Interlock Circuits - Significant count rate (source interlock) required on recorder for permission to operate rods
- Either period detector initiates setback on period shorter than 10 sec.

b. Log-N System (Channels III and IV)

Range - 1 watt to 1000 kw
Detectors - Compensated ion chambers (2)
Amplifiers - Combination Log-N and period amplifiers (2)
Data Presentation - Log N recorder for either channel with selector switch
- Period recorder for either channel with selector switch (shared with Start-up System)
- Period meter - driven from amplifier
- Log N meter - driven from amplifier

Auxiliary Circuits - CIC power supply for each channel

Interlock Circuits - Either period amplifier initiates scram on period shorter than 7 seconds (response time 8 msec)
- Period recorder initiates scram on period shorter than 7 seconds
- Log N recorder initiates scram on high (nominally 115%) power
- Log N recorder interrupts high voltage to fission chamber at 5% power
- CIC power supply failure sounds annunciator.

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c. Safety System (Channels V, VI, VII & VIII)

Range	- 50 kw to 10 Mw
Detectors	- Parallel circular plate ion chambers (4)
Amplifiers	- Composite safety amplifiers (2)
Data Presentation	- Power level recorder for any one of four channels with selector switch
	- Power level meter - driven from amplifier
Auxiliary Circuits	- None
Interlock Circuits	- Safety amplifier initiates scram on high (115%) power (response time - 8 msec)
	- Safety recorder initiates setback on high (110%) power

d. Linear Power System (Channel IX)

Range	- 5 watts to 10 Mw
Detectors	- Compensated ion chamber
Amplifiers	- Electrometer
Data Presentation	- Linear power level recorder driven from electrometer
	- Linear power level meter - driven from electrometer
Auxiliary Circuits	- Electrometer drives servo amplifier and magnetic amplifier for fine rod automatic power control sytem
	- CIC power supply
Interlock Circuits	- CIC power supply failure sounds annunciator

2. Process Instruments

The main gas loop and auxiliary systems are completely instrumented for operational control and experimental data. More than 250 individual instruments indicate (in over 100 cases) flow, pressure, temperature, and conductivity data and serve as interlocks in the control system.

In addition, the reactor is instrumented to provide the following data:

a. Mixed Mean Gas Outlet Temperature From Each Fuel Element

This data is displayed on a multipoint recorder and may be displayed on a 100 channel scanning oscilloscope on the console. Any single thermocouple (usually the hottest one) may be patched into a null balance indicator circuit which is equipped to sound an annunciator at a preselected temperature.

b. Individual fuel element flow is monitored at each of four locations in the core by means of a pitot-static system. This data is displayed on a meter in the control room.

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c. Four fuel elements are instrumented to provide temperatures of the individual fuel plates and gas temperatures in the gaps. The data from the 24 thermocouples on each element are displayed continuously on four 24-point recorcers. See Figure 16 for location of the thermocouples.

d. The reactor inlet and outlet ducts are instrumented to provide temperature data from which stress computations may be made. These data are presented on meters in the control room. These meters are equipped to sound an annunciator at a preselected safe temperature set somewhat above maximum operating temperature

3. Radioactivity Monitors

Ten radiation detectors are installed in the reactor area and at sensitive locations outside the building. The detectors monitor the activity at the gas and liquid waste tanks, filters, gas dryer, demineralizers, etc. and report the data to the control room. An individual meter is provided for each detector. The meters will trip an annunciator to sound on alarm at a pre-selected radiation level. This level will be set by the Asst. Project Manager and checked by the Health and Safety Officer.

In addition the air in the reactor room and control room is monitored continuously.

The gas circulating in the main coolant loop is monitored continuously for fission product activity. In case of an abnormal radiation signal, a monitoring scintillation counter will sample the gas effluent from each fuel element to determine the source of the activity.

A portal monitor and a hand and foot counter are installed at the normal exit from the reactor room.

F. REACTOR AUXILIARY SYSTEMS (See Figure 17)

1. Main Coolant Loop

The main coolant loop circulates nitrogen gas that contains 0.5% oxygen through the reactor at the rate of 63,000 lb/hr (nominal). The circulation of the gas is effected by a compressor with a variable speed drive. To avoid developing a high temperature compressor, a gas-to-water heat exchanger is used to cool the compressor inlet gas to 140°F by exchanging heat with a demineralized water intermediate cooling loop. This exchanger is a shell and tube type that is designed for 1250°F inlet gas and 10^5 hr life. Tubes are welded into the tube sheet and are ultrasonically and hydrostatically tested. The pressure balance across the tubes is such that nitrogen will leak out as long as the gas loop is pressurized in excess of 150 psig. The leakage of water into the gas loop when the loop is depressurized is considered to be improbable. Should it occur, the 150 psig water head would inject only a modest amount of water through a completely ruptured tube (1" OD, 12 BWG)

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which would be detected almost immediately by the electrode water detector in the main loop (LI-104).

2. Secondary Coolant Loop

This closed loop circulates demineralized water through the gas-to-water heat exchanger to cool the gas to 140°F. The secondary system dumps heat into the cooling tower heat sink system in a water-to-water exchanger. A surge drum is provided in the demineralized water system at the suction of the circulating pumps. This drum is pressured with nitrogen to keep the entire loop at a pressure sufficiently high to prevent any possibility of its boiling in the gas-to-water exchanger in the event of sudden load changes. In addition the operation of this loop in a pressurized condition provides a safety factor to compensate for the possibility of any eventual fouling of the heat transfer surfaces in the exchangers. A spare circulating pump is installed; either pump will circulate 1000 gpm in the loop.

3. Heat Sink

The heat sink for the facility is an induced draft cooling tower capable of cooling 2100 gpm of 105°F water to 75°F under conditions which prevail at the site. This system ultimately dumps all heat generated in the facility. The largest single load is the extraction of heat from the secondary coolant loop in the water-to-water heat exchanger. Either of the two circulating pumps is capable of circulating 2100 gpm. Provisions are made for pH and algae control as well as raw water make-up.

4. Heating and Ventilating System

Fresh filtered air is brought into the control building by a central heating and ventilating unit and distributed to all parts of the building except the control room and reactor building. This air is exhausted through strategically placed roof vents. A closed-cycle air conditioning unit circulates air of controlled humidity and temperature in the control room.

Fresh filtered air is brought into the reactor building by two space heaters. This air is exhausted by a fan to the building ventilation stack. Air is changed in the piping and equipment pits about once per minute and in the balance of the building about six times per hour.

This arrangement results in the control building and control room being at slightly higher pressure than the reactor building. For this reason, any air transfer will be in the direction of the potentially contaminated area. This desirable balance will be checked before start-up and adjustments made as necessary. Rechecks will be made at regular intervals after start-up.

5. Radioactive Waste Disposal System

The facility is provided with equipment and piping to assure

complete control of liquid radioactive waste. All floor drains and equipment drains in potentially contaminated areas terminate in a sump in the bottom of the heater pit. From this sump, the liquid may be pumped to (a) the percolation field, with or without dilution, or (b) to either of two 10,000 gal liquid waste tanks.

Provision is made for decontaminating the reactor and main coolant loop by circulating any one of a number of deontaminating solutions. These solutions may also be pumped either to the percolation field or to the waste tanks.

Waste in the two 10,000 gal tanks may be pumped to the percolation fields with or without dilution, or may be precipitated or permitted to settle in the tanks. Provision is made for decanting and slurry handling so that ultimate disposal may be effected.

Radioactive resin is sluiced into drums, decanted, sealed and disposed of by normal means. Radioactive gas is collected in a 10,000 cu ft waste gas holder from which point it may be exhausted, with dilution as necessary, to the exhaust stack. This, as well as all other disposal operations, will be conducted under specific procedures to assure compliance with NRTS regulations and to avoid creating a hazardous condition.

6. Electrical Distribution System

Power is supplied to the site from the 13.8 kv NRTS distribution network. This is transformed in the 1000 kva substation to 480 v for site distribution. Non-critical loads are bussed in Motor Control Center #1 from the normal power supply. Motor Control Center #2 which supplies the instrumentation, security lighting, space heaters and certain sensitive motors is powered from the normal supply. An automatic transfer switch connects this motor control center to the emergency bus in case of failure of the normal supply. Motor Control Center #3 supplies the emergency equipment which is required for safe shutdown of the reactor. This control center is normally connected to the emergency bus but a transfer switch provides normal power if the emergency power fails. The emergency bus is energized by a 300 kw, 480 v diesel-powered generator.

7. Building

The main structures at the site are the adjacent and connected control building and reactor building. The former is a one-story concrete block structure which houses the offices, laboratories and the control room. The reactor building is a 25-ft-high steel and asbestos cement board structure which accommodates the reactor and most of the auxiliary equipment.

The control room is separated from the reactor pool area by a distance of approximately 30 ft and also by an 18 in. concrete wall.

In addition the roof and walls of this room are 1-ft-thick concrete. The counting room is completely enclosed by 1-ft-thick concrete walls and roof and is located some 50 ft further away from the reactor pool.

A plan view of these two buildings is shown in Figure 4 along with the location of the major pieces of equipment.

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IV. OPERATION

A. PERSONNEL

The reactor operating staff will consist of twenty-eight persons as shown on the organization chart in Figure 18. In selecting personnel to fill the organization, consideration was given to experience, maturity, competence, and personality. The following sections briefly describe the functional responsibilities and authorities of the key personnel.

1. Assistant Project Manager

This person is the ranking supervisor in the organization and is charged with overall responsibility for the safety of personnel and equipment, and for executing the test program. He formulates or reviews and approves all operating and experimental procedures, as well as safety procedures and regulations. He maintains liaison with, and serves as a contact for the USAEC-IDO and Aerojet-General Corporation.

2. Operating Superintendent

This person is ranking operating supervisor on the GCRE I and is charged with responsibility for safely operating the reactor and executing the test program. He assists in reviewing operating and experimental procedures and safety rules and regulations. He insures that the operation plan and experimental program are implemented and coordinates maintenance work.

3. Health and Safety Officer

This staff member is charged with the responsibility for auditing the radiological and industrial safety performance. He reviews and approves all operating and experimental procedures and all safety rules and regulations. He maintains and/or reviews all radiological safety, exposure and survey data. He recommends and/or consults with supervisory personnel on problems of handling of radioactive materials, decontamination, and elimination of radioactive or industrial safety hazards.

4. Senior Nuclear Engineer

This staff member is charged with the responsibility for auditing the technical performance of the organization. He reviews and approves all operational and experimental procedures and equipment for sound neutronic and engineering design and safety. He consults with supervisory personnel on operational and experimental problems and provides neutronic and engineering recommendations. He is responsible for observing experiments to insure that proper data are collected and recommends modifications to experimental procedures where indicated.

5. Shift Supervisor

This person is responsible for adhering to operating and experimental procedures, safety rules and regulations, and daily instructions. He is responsible for observing operations and detecting incipient or existing non-standard conditions and for taking the necessary corrective action.

B. PROCEDURES

All aspects of the GCRE-I operation will be covered by detailed procedures. Compliance with these procedures will be mandatory.

The procedural system is comprised of the following:

1. Technical Manual

A complete and detailed narrative description of the process and equipment and of the contemplated methods of operation. This volume will serve as a reference source and as a training aid.

2. Technical Standards (ANTS)

A compilation of standards which outline the basic rules and limitations of operations. These standards describe the extremes of temperature, pressure, flow rates, etc. imposed by the design, as well as the procedural and administrative requirements for safe operation. They serve as the basis for preparing procedures and establish the guide lines within which the experimental procedures are drawn.

The ANTS are prepared by the operating group and reviewed and approved by the Aerojet Supervising Representative and the AEC Contracting Officer. Selected ANTS are presented in Appendix E.

3. Standard Operating Procedures (ANSOP)

These are the detailed step-by-step operating procedures that cover all phases of the activities of the operating organization. A partial index of ANSOP is included in Appendix F to demonstrate the scope of these procedures. Of particular interest to this report are Sections 2 (Radiological Safety), 3 (Safety), 5 (Reactor Operation), and 6 (Experimental Operation). Selected ANSOP that are pertinent to this report are also included in Appendix F and are referred to in the text of this report.

The ANSOP are prepared by the operating group and are reviewed by a ten member Procedure Review Board prior to their being submitted to the Aerojet Supervising Representative and the AEC Contracting Officer for approval. Revisions to the procedures follow the same channel for authorization except that revisions initiated at the site are processed by a four member board that consists of the; Assistant Project Manager, Operating Superintendent, Health & Safety Officer, and Nuclear Engineer.

4. Standard Operating Logs (ANSOL)

This category includes:

a. The check lists that are used in field to implement the procedures. These are extracted from the ANSOP and present only enough information to guide the operator in perfoming the procedure. This form includes a space for the operator to initial which indicates that the step has been completed.

b. The data sheets upon which the operating data of the process are recorded.

c. The log books which give a narrative description of the process and support the data compilations of (2) above.

ANSOL are developed by the operating group and are approved by the Assistant Project Manager.

C. TRAINING

The operating personnel on the GCRE I will be thoroughly trained in the operating and safety aspects of the work prior to their reporting to the job site. This training will be accomplished by an organized program of instruction and testing. The assignment of key personnel to the operating group as much as a year in advance of the contemplated start-up date has permitted a thorough study and analysis of the operation. Participation in such activities as design review and liaison, construction liaison, and procedure preparation and review has permitted these persons to develop a detailed knowledge and understanding of the equipment and how it should operate.

All operating and maintenance personnel on the job will complete the training program prior to the start-up of the reactor. After operation has started, on-the-job training will include periodic and unannounced drills of reactor emergency procedures as well as detailed reviews of procedure and equipment revisions, and experimental procedures. The training program will be revised as required to keep it current with process revisions and will be used to instruct new personnel as required.

D. PERSONNEL SAFETY

1. Non-Radiological Hazards

A completely integrated safety program has been developed for the operation as indicated in ANSOP Section 3. This program complies with the Standard Health and Safety Requirements, Issue No. 3-1957 published by the Idaho Operations Office, USAEC. It includes: (a) an inspection phase, wherein attention is devoted to handrails, rotating equipment guards, stairways, tripping, falling or bumping hazards, emergency exits, etc. and (b) an operational phase which provides for regular safety meetings, housekeeping, safety and fire inspections, fire drills, and the development and implementation of safety rules and regulations.

First aid facilities and trained personnel will be available at the site for treating minor injuries and rendering immediate first aid assistance in the case of serious accident. The crew will look to the established medical organization at the NRTS for professional assistance when required.

Fire fighting equipment and procedures will be available and all personnel will be trained in emergency action and in fighting small fires. Here again, assistance will be available from the trained professional fire fighting organization at NRTS if required.

2. Radiological Hazards

The primary operating plan of the GCRE-I with respect to radiological hazards is that no person shall receive any radiation exposure that is not required by and justified for the performance of the job. Such exposure as is necessary, will be controlled and monitored by accepted and proven means so that permanent records can be maintained on the radiation history of each individual. Maximum permissible exposures are established which are at least as safe as those laid down in the most current AEC regulations. Radiation areas will be clearly marked and admission to such areas will be granted only after the exposure levels are known and time limits are established. Persons entering radiation areas will do so under a control system which specifies their conduct in the area.

The backbone of the radiological safety program is the Special Work Permit (ANSOL 2250). This form is, in essence, a contract signed by the person who is to perform the work and the person responsible for his safety. It covers the following points:

a. Description of the work to be performed, including the area and equipment, and sufficient details to permit an accurate evaluation of the hazards.

b. Estimate of the maximum exposure rate to be encountered. This is based on previous surveys and on information presented by the remote monitoring system.

c. Estimate of the maximum time limit to perform the work. This is based on the maximum permissable exposure limit and item (a) above.

d. Specification of any special conditions to be observed in the prosecution of the work.

e. Specification of all protective equipment to be used on the job, including special dosimeters if required.

f. Provision for completing the record of the exposure accumulated while on the job.

g. Signatures of the responsible supervisor and the person doing the work. This indicates that the first has evaluated the hazard, and that the second understands the hazard and how he shall protect himself in prosecuting the job.

The Special Work Permit is supplemented by the Exposure Record Card (ANSOL 2251) on which employee keeps a record of his accumulated exposure to prevent overrunning the maximum permissable daily or weekly exposure.

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A Master Emergency Plan and evacuation procedures have been developed and periodic drills in these procedures will be conducted.

E. PROGRAM

1. Pre-Start-Up Testing

The pre-start-up testing of the facility will be accomplished in several phases:

a. Cold Performance Testing

In this phase, all reactor support equipment will be tested to ascertain that it performs in accordance with specifications. After the individual components are proven, the various systems will be operated in a cold (both thermally and radioactively) condition. Procedures have been developed for this work and the program generally is established. The systems included in this phase are the utility water, cooling water, utility air, instrument air, fuel oil, liquified petroleum gas, demineralized water (secondary coolant loop), waste nitrogen, liquid waste and loop cleaning. Coincident with this testing, certain specialized calibrations, e.g. reactor flooding system, will be completed.

b. Instrument Performance Testing and Calibration

Concurrent with phase a, all instruments in the facility will be checked, insofar as possible, by the use of artifical signals where necessary. Temperature sensing elements will be exposed to heat sources and the indicating, recording and controlling circuits proven out and adjusted. Dead-weight testers will be used to actuate pressure sensing systems. Artifical signals will be applied to flow metering and neutronic systems for proof testing and calibration. Alarm, interlock, setback and scram actuators will be set. Control valves will be adjusted and relief valves bench tested. Necessary repairs and adjustments will be completed to make the facility ready for phase d.

c. Cold Testing of Main Coolant Loop

During this phase, the main (gas) coolant loop will be pressurized and circulated in the cold condition. The lubricating oil, seal water, speed control, and emergency drive systems on the main compressor will be proven out. The make-up and bleed system will be checked and adjusted. The emergency blowdown system, the steam cooling system and the shutdown cooling system will be tested and adjusted as necessary. Initial circulation will be done with filters in the line to protect the reactor and to establish the cleanliness of the system.

d. Hot Testing of the Facility

Using the oil-fired heater as a heat source, several circulating runs will be made under increasingly severe conditions of temperature and pressure to proof-test the design and installation of the system. The most severe run will be at 200 psig and 1200°F. During these tests, strain gage determination will be made at critical points in the system and stress calculations will be performed to check the design conditions against the findings. The most severe test will be repeated at intervals no longer than six months during operation.

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e. Control System Checking

The control system will be checked in detail to be certain that all neutronic, electronic, electrical and mechanical components function properly. Rods will be cycled, interlocks and scram circuits checked and necessary adjustment made.

f. Final System Check

When all the above checks have been completed, the entire system will be operated under simulated operating conditions (with dummy fuel elements in the reactor) for a 24 hour period. During this period the operating controls, i.e., pressure, temperature, flow, etc., will be exercised. The control rods will be operated and all instrumentation checked to assure system readiness for the initial critical experiment.

2. Initial Critical and Zero Power Testing

The initial critical experiment will be conducted in accordance with ANSOP 9200 in Appendix F. The program provides for determining the cold minimum critical loading both wet and dry, for establishing the worth of additional fuel elements, and finally, for attaining the reference operating core loading with its associated critical rod configuration. This work will be performed with the reactor open and the pool partially filled.

Following this work, a series of zero power experiments will be conducted which include rod worth determinations and rod calibration curves, the flooding experiment, coefficient measurements, zero power flux mapping, and neutronic instrument evaluation, calibration and adjustment. The temperature coefficient will be measured by observing the reactivity effects of oscillating the coolant temperature about 30°F, on a 10 second cycle with the reactor at low power. This technique permits the fuel to reach equilibrium at the new temperature, but does not result in an increase in moderator temperature. The determination will be made at several points between ambient temperature and 1200°F.

3. Power Ascension Testing

At the conclusion of the zero power testing, the power ascension and the associated test program will be initiated. At very low power, the effects on reactivity of altering gas pressure, temperature, and flow will be determined. During the power ascension, radiation surveys, gaseous waste analysis and general system performance evaluations will be conducted.

4. Fuel Element Testing and Systems Evaluation

After proving that the system is capable of operating at the reference conditions and after performing the tests and experiments necessary to establish the neutronic and mechanical characteristics of the facility, sustained operation at reference power will be initiated. During this phase, the fuel element testing program and certain systems evaluations will be conducted. Reorientation

and relocation of the instrumented fuel elements will be undertaken to provide complete data on fuel element environment and performance. Improved fuel element designs will be tested by installing a limited number of elements in the core and evaluating their mechanical and neutronic characterisitcs. Tests will be conducted to evaluate the effects of rapid transients in flow and coolant temperature on reactivity.

The following fuel element experiments are planned in the GCRE-I:

a. Calibration of the fuel element instrumentation, orifices and determination of heat loss from the core.

The calibration of all the fuel element thermocouples and pitot tubes will be made by pumping heated gas at various temperatures and flow rates through the non-critical reactor. Because the reactor is not generating heat, the decrease in enthalpy of the gas flowing through the reactor will be equal to the heat loss through the pressure tubes and the temperature of the gas flowing through the core will decrease uniformly as it passes through the core.

The experiment will cover flow rates from 20 to 120% of normal, and gas temperatures from 100 to 1200°F. For each condition, the temperatures of all the instrumented fuel elements, all the reactor thermocouples (including the gas inlet and gas outlet temperatures) and the water within the lead shield will be recorded. The flow rate will be recorded at the main flow meter and at the four pitot-static sensing locations in the reactor. The pressure record will include the barometric pressure, system pressure, and the pressure drop across the core.

The calibration of thermocouples will be accomplished by comparison of all the temperature readings from the thermocouples which are expected to give equal readings at equal distances from the inlet end. The heat loss from the reactor will be determined by making a heat balance across the reactor and by plotting the heat loss through the water against the average temperature of the gas less the temperature of the water. The flow rate through the core will be plotted against the pressure drop across the core at each temperature level. The flow through selected individual elements will be plotted against the pressure drop across the core and these data compared with the orifice calibration curve obtained in the laboratory.

b. Effect of Rod Position on Flux Distribution

The power distribution in the core for the various rod configurations that will be used in GCRE-I will be studied. This measurement will also include detail intracell distribution determinations on selected elements. This experiment will be performed during the zero power tests, using monitor wires which are normalized against uranium foil.

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c. Burnable Poison Study

The effect of burnable poison depletion on the flux shift within the element will be measured by making intracell flux traverses at intervals of 500, 1000, 2000, and 3000 hr. The intracell measurements made during the zero power experiment will complete the data to give the burnable poison effect from zero to 3000 hr intervals. This experiment will be performed by the use of monitor wire in the fuel element gaps.

d. Heat Transfer Data at Steady-State

During the steady-state operation of the GCRE-I, essentially continuous measurements will be made and recorded of the temperature and flow on all the instrumented elements. From these data, it is expected that the heat transfer data will be obtained by comparing the experimental results to the predicted values.

e. Hot-Spot Data at Steady-State

Because the experiments under Paragraph d (above) will be conducted throughout the lifetime of the core, the change in heat transfer characterisitcs may be used to indicate gap distortions, i.e.,the development of hot spots. The hot spots which result from the fuel element manufacturing tolerances will be established by analysis of the initial data and subsequent gap distortions will be detected by temperature shifts. Hot cell examination after removal of the element will confirm the deductions.

f. Temperature Transients Due to Rod Motion

Instrumented elements will be positioned in the reactor to observe the temperature transients introduced by rod motion. The temperature variations are studied on the oscilloscope. The variations resulting from normal operating rod motion as well as ABLE and BAKER scrams will be investigated.

g. After-Heat Measurement

The after-heat will be measured to establish the length of time full flow circulation is required after a reactor shutdown. This experiment will be performed by discontinuing the compressor operation at various times after shutdown and measuring the equilibrium temperature attained by the fuel elements. The initial test will be made after sufficient time has elapsed, as determined by calculation, to avoid damage to the element. The shortest time interval between shutdown and cessation of full flow circulation will be determined experimentally as the point which results in fuel element temperatures less than 1200°F.

h. Diffusion Measurement

During operation of the GCRE-I, the main coolant gas will be sampled periodically and analyzed radiographically for evidence of diffusion through

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the fuel element cladding. The sample will be taken on a charcoal filter product or similar absorber to increase the sensitivity for determination of fission products. This will be in addition to the continuous monitoring for gross activity described earlier.

i. Part Load Operation

Operation of the GCRE-I at flow rates in the transition range (turbulant to laminar flow) will be performed to study the effects of this mode of operation on the heat transfer characteristics. These data are expected to be useful in the AGCR operation. The test will be performed by slowly reducing the flow rate while the reactor power is reduced accordingly so that the maximum temperature of the element shall not exceed that attained during the normal operation. During the part of the test which is expected to be near the transition range, the highest reading thermocouples will be connected to the oscillograph to study any oscillation which may develop. The flow rate will then be decreased further and the temperatures observed closely. The flow rate will then be slowly increased and the temperature data at the various flow rates recorded to detect hysteresis.

j. Shutdown Cooling Experiment

The purpose of this experiment will be to check the calculations which indicate that no auxiliary cooling system is required in case of complete flow stoppage in the GCRE-I. The results of this experiment will be useful for planning the emergency cooling for the AGCR. To check the validity of the calculation, the experiment will be conducted by starting with a known safe shutdown cooling flow and reducing this to zero, meanwhile monitoring fuel element temperatures to prevent a maximum indication in excess of 500°F. Having established this point, the test will be re-run from nominal operating conditions. The coolant flow will be diverted around the core so that a no-flow case through the reactor can be simulated and the reactor simultaneously scrammed. The temperature transient will be watched on the oscillograph. The interval between flow stoppage and scram may be increased slowly as determined by the maximum temperature attained during the experiment. In any event, the compressor will be running so that coolant flow can be quickly re-established.

F. NORMAL OPERATION

As previously stated, all aspects of the operation of the GCRE-I will conform to authorized procedures. To implement this program, a complete set of "normal" operating procedures is being developed. As experimental procedures are written, it is anticipated that broad areas can be covered by reference to the applicable normal procedures. This approach will reduce the effort required for experimental procedures.

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Under the normal procedures, the reactor start-up will be performed in accordance with the following procedures. The asterisk indicates that this procedure is included in Appendix F.

1. ANSOP 5704* Pre-Start-Up Check List

This procedure outlines the checks to be made of the auxiliary systems to assure that conditions are satisfactory for reactor start-up. It also includes the initial pressurization of the gas loop and start-up of the main compressor, instrument checks (ANSOP 5705, 5709 and 5713*) start-up of the heater, safety circuit checks, (ANSOP 5707*) rod operability checks (ANSOP 5710) a check of the remote area radiation monitoring system and a check of all valves in the system (ANSOP 5708*).

2. ANSOP 5702* Normal Start-Up

This procedure covers the pre-start-up adjustments, the rod withdrawal, attainment of criticality, and power rise to 150 watts. The rod withdrawal sequence is specified and the rate of reactivity addition controlled.

3. ANSOP 5706* Increasing Power

This procedure outlines the course of action for raising the power level to the prescribed operating point.

During normal operation, ANSOP 5703* specifies the rules to be followed and ANSOP 5712 lists the routine reports to be completed. In case of a scram, setback or annunciator, ANSOP's 5805 through 5877 describe the immediate and subsequent action to be taken.

Reactor shutdown is outlined in ANSOP 5715*. Cooling following shutdown is covered in ANSOP 5717* and shutdown neutron monitoring in ANSOP 5616*.

Reactor flooding, with safeguards against reaching a critical state is outlined in ANSOP 5718* and fuel element handling is covered in ANSOP 5721*. Reactor drying will be accomplished initially by using open loop circulation of air heated to 800°F and subsequently by closed loop circulation of nitrogen gas. Adequate procedures will be provided to prevent steam formation in the fuel element insulation. Water will be condensed from the circulating closed loop nitrogen gas by cooling it to 140°F in the heat exchanger. The condensate will bleed off at the provided drains when system pressure is reduced for purging with dry nitrogen prior to reactor start-up. A ruptured fuel element will be removed by withdrawing the element from the core into a light metal container that can be remotely sealed. Having thus contained the fission product activity, arrangements will be made to transfer the element to a hot cell facility for inspection.

V. HAZARDS EVALUATION

A. INTRODUCTION

The potential hazards associated with operating and testing the GCRE-I will be discussed in light of two major considerations:

1. The possible mechanisms for introducing uncontrolled reactivity into the reactor core. These will include a definition of the maximum credible accident, and other occurrences which could result in physical damage to the reactor.

2. The consequences of the maximum credible accident, which will include energy release, mechanisms of shutdown, and radiation hazards resulting from the energy release.

B. MECHANISMS FOR CREATING POTENTIAL HAZARDS AND THE MAXIMUM CREDIBLE ACCIDENT

1. General Considerations.

In designing and evaluating the GCRE-I system, the primary safety considerations that were included are:

The reactor temperature coefficient, either prompt or steady state, is not sufficiently large to have a significant effect on reactivity under any condition. In an uncontrolled excursion, the primary shutdown mechanism is the gross deformation of the fuel elements.

The reactor is undermoderated. Flooding the coolant passages and the voids in the insulation with water will result in an increased reactivity of approximately 10.8%.

The reactor control system is designed so that potential hazards from reactor excursions are limited by the following factors:

The excess reactivity that is available to the operator, when the core and the insulation is dry, is 2.4%.

The maximum rate of reactivity input from the control system is less than 0.03%/sec, and will not bring the reactor to a prompt critical condition before shutdown.

The entire control system has been designed so that no single equipment failure or operational error can result in an unsafe condition.

Various possible causes of reactor excursions are discussed below. An attempt has been made to evaluate the probability of occurrence of these conditions. The consequences of these accidents are discussed in Section VI, D. Figure 20 shows the excess reactivity which is reached by inserting reactivity at a given rate.

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2. Control System Failure

To minimize the probability and consequences of operational errors, interlocks are provided that prevent the extraction of safety and shutdown rods unless certain conditions are satisfied. With the shim rods inserted, the extraction of the shutdown and safety rods will not make the reactor critical, as shown in Figure 10.

All safety circuits and mechanisms are operated in parallel, with completely independent actuation. Thus, the failure of a single item will not jeopardize the safe shutdown of the system, and a hypotehtical accident mechanism must be postulated on multiple equipment failures and operational errors. The worst accident from these multiple causes would result from the following maloperations:

a. All scram and setback systems inoperative.

b. Control-rod interlocks inoperative.

c. Circuitry modified to allow power to be supplied simultaneously to all control rod drive mechanisms.

d. The reactivity is increased through criticality and for 30 sec additional, with all control rods being simultaneously withdrawn at their maximum rates, giving a total ramp input of reactivity of 0.026%/sec.

e. Multiplication not checked by operator during withdrawal.

The simultaneous occurrence of these events is not considered to be credible. A plot of power level vs time for this situation is shown in Figure 19. At a power level of 2 megawatts, the reactor is on a period of 0.3 sec. Shutdown occurs at a power level of about 650 megawatts, and at a reactor period of 0.15 sec, due to fuel element melting and partial vaporization as described in Section VI and Appendix A.

The maximum possible accident from equipment failure and operator errors must include the effect of the absence of a neutron source in the reactor in addition to the conditions listed above. Under these conditions, the excess reactivity could increase to the total available value of 2.4% before a chain reaction started. This would result in a reactor period of 4 msec. Because a Sb-Be neutron source will be permanently attached, its presence will be assumed in all credible situations.

3. Deliberate Flooding of Core

Flooding the coolant passages with water adds a total of 7.9% reactivity. Saturation of the insulation material with water provides an additional 2.9%. Deliberate flooding will be done only when the shutdown control rods are inserted, rendering the system sufficiently subcritical to provide complete safety. This safety is provided by any six of the eight safety and shutdown rods.

The core must be flooded prior to removing the top cover plate of the reactor for fuel element access. This deliberate flooding will be done at a rate of 1 gpm, giving a reactivity increase of less than 0.02%/sec. This rate is inherently safe, and cannot cause a reactor excursion even with complete control system failure. Water level and reactor multiplication will be monitored continuously during this procedure. The maximum core flooding rate is limited to less than 2 gpm by the pipe size and by the water pressure available in the flooding system. The resulting maximum reactivity increase is .03%/sec. The worst accident associated with deliberate reactor flooding would result from the coincident occurrence of the following maloperations:

a. Reactor shutdown with three or more shutdown rods not inserted, or with the equivalent reactivity that is normally not available.

b. Either the shutdown rod position monitoring system must be inoperative, or the flooding procedure ignored.

c. Multiplication not checked during flooding period. This accident would be similar to that caused by the control system maloperation, shown in Figure 19,

4. Accidental Flooding of Core

A large rupture in the plenums or in the large pipes within the pool could result in rapid flooding of the reactor coolant passages. While such a large rupture appears to be unlikely, it is credible that it could occur as the result of a pressure shock, a severe stress from thermal cycling, or external factors such as an earthquake. If the rupture occurs while the reactor is shut down, with the shutdown rods inserted, no excursion will result.

The effect of rapidly flooding a hot, critical reactor and of rapidly flooding a cold, near critical reactor, are considered below:

a. Rapid Flooding of a Hot, Critical Reactor

Flooding the reactor when it is at operating temperature will generate a large amount of steam in the coolant passages. This steam will significantly increase the time required for flooding. Under these conditions, the coolant channels cannot be flooded in less than one second. This is sufficient to allow the scram system to function. While the core structure and fuel elements would be severely damaged from steam explosions and from thermal shock, no excursion will occur and the release of fission products from broken fuel elements will be small. Because the same physical action which ruptured the core (earthquake, etc.) might also disable the fast scram system in some manner, it is believed credible that flooding might occur and cause an uncontrolled excursion. A one second flooding of the coolant passages inserts an average ramp increase of reactivity of 8%/sec from the critical condition. This is equivalent to a 1.5% step increase in reactivity (Figure 20) in which the reactor power increases through nine periods before the reactor is destroyed.

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b. Rapid Flooding of a Cold, Near Critical Reactor

If the core structure or underwater gas piping sustained a large rupture while the reactor was in a near critical condition, the reactor could be flooded through criticality. This excursion could be more severe than the hot, critical accident because the flooding would not be impeded by steam formation in the gas passages. Also, because the initial reactor power level is about 10^8 below full power, the reactivity could increase to a much higher value before the various shutdown mechanisms (based on temperature) begin to operate (Figure 20).

It is considered credible that the reactor could be cold and near critical (although this will never be done in normal procedures) and that some physical action could rupture the pressure vessel and disable the fast scram system. This will allow the core to be flooded in a time as short as 1/2 second. This ramp increase in reactivity of 16%/sec from a sub-critical condition is equivalent to a 3.0% step increase. This excursion represents the maximum credible accident for this reactor. The operation of the scram system normally would prevent the occurrence of a hazardous excursion from this flooding rate.

One-half second is considered to be the minimum credible flooding time. However, if water under a 25 ft head were introduced into all coolant tubes simultaneously and if this pressure were maintained without back pressure of any kind, it would be possible to flood these tubes at 0.17 sec. No mechanism or accident is postulated which would produce this result.* If it were to happen when the reactor was near critical, the resulting ramp increase of reactivity of 80%/sec would be equivalent to a step increase of approximately 5.5%. This would cause a reactor period of 1 msec which represents the maximum possible accident. This flooding accident could not be controlled by the scram system.

5. External Physical Damage

Certain external factors, such as a severe earthquake, a bomb, an airplane crash, or sabotage are capable of damaging the reactor system in a way that would cause a condition hazardous to the surrounding area. In addition to causing an excursion, such an event might severely damage the reactor building and allow a high rate of contaminated leakage.

Several possible types of accidents are listed below:

(a) An event that would cause a large rupture in the plenums or gas passages. This eventuality was covered in Sec. 4, " Accidental Flooding of Core", and is the cause of the maximum credible accident.

(b) An event that would remove all control rods from the reactor. The reactor would then be made 2% supercritical, which would result in a 4 msec period. The same event also would certainly rupture the core structure and cause a flooding accident with the resulting 3% equivalent step.

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* A pressure of 70 psig will be maintained in the system during cold start up. The nitrogen bottles will provide a flow of 38,000 standard cubic feet per hour at 250 psig into the system.

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(c) An event that would change the fuel element configuration. The most likely event is one in which a force external to the reactor moves some of the fuel elements toward the center. The resultant decrease in fuel plate spacing would cause a decrease in reactivity.

(d) An event that in some manner increased the fuel element spacing, more or less uniformly and allowed water to fill all voids, could result in an increase in reactivity of up to 3%. This event is not considered credible.

6. Addition of Fuel

During the normal fuel loading operations the reactor core and coolant passages will be completely filled with water. When flooded, the reactor has an inherently safe shutdown mechanism that is caused by steam formation adjacent to the fuel elements. An investigation that compared the GCRE-I flooded core to the SPERT core revealed many similarities in: fuel element thickness and spacing, core size, neutronic characteristics, and in thermal conductivity between the fuel and the water. Thus it may be said, qualitatively, that the flooded core possesses the inherent shutdown mechanisms found in SPERT.

Replacement fuel elements normally will be placed only at the core periphery. The worth of one new element added to the periphery of the near critical reactor is 0.40%. The most reactive element which could be added is one placed in the center position of the core lattice. If the reactor were in a near critical condition with the center tube empty, one fuel element added in this position would increase the reactivity by over 2%. All possible safeguards will be applied during fuel element loading operations to prevent such an occurrence. However if gross human error and equipment malfunction would cause criticality, an excursion somewhat similar to those of SPERT would result. Such a condition would be disastrous to the fuel handlers but would not create a safety hazard outside the immediate area.

7. Sabotage

The primary safeguard against sabotage must be placed on the security system, since it must be assumed that a well-informed saboteur could cause an excursion regardless of design.

The only method that is available to a saboteur without physical access to the submerged reactor core is simultaneously to withdraw the control rods and to flood the core at the maximum rate (2 gpm) as described in sections 2 and 3. It is not possible by these methods to place the reactor on a prompt period.

While much more severe accidents are possible, physical access to the core is required in order to obtain very rapid flooding of the coolant passages or to remove the neutron source.

8. Dynamic Stability

Analog studies were performed at the Oak Ridge National Laboratory Analog Facility which simulated the GCRE-I control system. The results of these studies

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are reported in ORNL Report CF-58-7-2. A response curve typical of those developed by the study is shown in Figure 21. In this case a 0.05% ΔK/K step increase in reactivity was imposed on the system at a power level of 2.35 Mw. The temperature coefficient for this case was assumed to be + 1 x 10^{-5} / °F. The power reaches a maximum of 2.65 Mw before recovery.

From the GCRE-I stability studies that were conducted, it may be concluded that the present maximum regulating rod speed of 0.01% Δ K/K per sec and total rod worth of 0.4% Δ K/K are capable of controlling quite severe perturbations in reactivity, even for a system possessing strong positive fuel temperature coefficients.

9. Fluid Temperature Changes

The moderator temperature coefficient varies between +1.0 x 10^{-4} Δ K/K per °C at 20°C and -0.3 * 10^{-4} Δ K/K per °C at 80°C. The total reactivity increase due to moderator temperature rise as the power is increased from zero to full power is about 0.15%. This temperature rise requires about 100 sec and can be controlled by the regulating system.

The reactivity that is controlled by the nitrogen in the coolant channels is about 0.2%. This figure represents the maximum instability which can be caused by pressure or temperature changes in the coolant gas. Even under the improbable situation of very rapid changes, operation of the regulating or setback rod will prevent the occurrence of any dangerous condition.

10. Loss of Coolant Flow

The interruption of the coolant flow from any reason will cause the reactor to be shut down immediately. The heat generation a few seconds after a scram is less than the normal heat lost to the water through the insulation. In addition, there exists a thermal coupling between the fuel elements and moderator, via radiation and conduction, to prevent severe overheating of the elements if the reactor is scrammed immediately following the loss of coolant flow. If the reactor is scrammed within one second after the loss of the coolant flow, the maximum fuel element temperature rise is about 200°F. The nominal fuel element temperatures for this case are shown in Figure G-2, Appendix G. The hot spot temperatures would be higher (c.f. Appendix G). The maximum temperature occurs in the inner fuel tube about five minutes after a scram. A gross fuel element meltdown, due to the loss of coolant, can occur only if it is accompanied by a complete control system failure so that the reactor is not scrammed.

In the event of line power failure, the fired-gas heater and reactor will be shut down and the main coolant gas compressor will be driven at half-speed on diesel-electric power. This shutdown will be gradual, and will reduce reactor temperatures to 300°F in approximately one hour. At that time, the operator will normally reduce the pressure of the system to atmospheric and turn on the emergency shutdown blower for the desired shutdown period or until the reactor is flooded for maintenance or inspection purposes.

If the fuel elements should melt, the reactor core structure will contain the molten metal, and the evolved fission products will be contained within the loop.

11. Power Failure

If the line power should fail, a continuously operating 300-kw generator will supply the necessary emergency power for the GCRE site. The reactor will be scrammed and the shutdown cooling system will be operated as described previously. In the improbable event of a simultaneous failure in line power and in the 300-kw emergency power system, the nitrogen blowdown cooling system will function for several minutes to prevent any possible excessive fuel element temperature rise.

12. Discussion of the Maximum Credible Accident

An inherent characteristic of compact heterogeneous gas-cooled reactors is that the filling of the gas passages with water will result in a large increase in reactivity. That is, in order to attain criticality in a small size, the reactors must have a large uranium loading and a high neutron leakage. Filling the gas passages with water decreases the neutron migration area and leakage but does not significantly decrease the neutron utilization. This inherent characteristic was realized early in the study phase on gas-cooled reactors and the entire design effort has been made with this feature in mind. Wherever possible, special precautions have been taken to prevent or at least to limit the rate of water flooding the gas passages. In addition, the control rods were designed to over-ride the most rapid flooding rate conceivable.

Even with these precautions, water flooding of the gas passages still represents the greatest potential hazard in that it is the most rapid method of introducing positive reactivity into the core. For the purpose of this analysis, the maximum credible accident for GCRE assumes the simultaneous occurrence of rapid flooding and a complete control rod failure while the reactor is in a condition that represents the most pessimistic combination of circumstances.

The water head that is available for flooding the GCRE-I is 25 ft which is the depth of the water above the reactor in the pool. If this head of water is applied to a GCRE-I fuel tube and if this head is maintained, the coolant passages will be flooded in 170 msec. Because of water turbulence, entrance losses, and impedances in the plenums it is not considered credible that this head could be maintained on any individual element. In addition, there are 61 individual fuel tubes and in order to gain large amounts of reactivity it would be necessary to flood all (or nearly all) of these tubes. For these reasons the minimum conceivable time to flood the gas passages of GCRE-I is believed to be in excess of 1/2 second. The reactivity increase that would result from flooding all of the coolant passages is 7.9%. The time necessary to saturate the insulation with water is about 5 sec and is too slow to have a significant effect upon the maximum flooding accident.

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In order to flood the reactor at this rate it would be necessary to suddenly cause a large opening in the underwater portions of the reactor gas passages. This opening would necessarily have to be several square feet in area in order to achieve the rapid flooding. This could only occur due to the complete removal or the gross physical failure of the inlet or outlet plenum. Because of the large safety factors used in the design of the reactor pressure vessel, failure is extremely improbable. The most probable cause of failure would be dropping some very heavy object on the vessel. Operating procedures specify that the reactor bridge shall be positioned and locked above the reactor during operation. This will minimize the probability of this occurrence. The vessel has been designed to prevent damage to the reactor pressure vessel due to earthquake or similar external shocks. The control system has been designed to prevent a dangerous excursion in the event of a flooding accident, however improbable. Therefore, in order for such an accident to become hazardous it must be accompanied by a control system failure and must take place while the reactor is in a cold but nearly critical condition. The excursion energy resulting from this accident has been calculated to be 380 megajoules. This will be accompanied by a chemical energy release of about 100 megajoules due to metal-water reactions. The accident will cause about 25% of the fuel elements to be vaporized and most of the remainder to be melted.

It may be argued that the selection of 1/2 second as being the minimum credible flooding time is somewhat arbitrary, or that the selection of an average rate of reactivity during flooding of 16% per second is not a true representation of the actual case. This is admittedly so because of the many uncertainties in arriving at an accurate description of the flooding process immediately following an accident which ruptures the inlet or outlet plenums. Nevertheless it can be shown that the accident analysis is relatively insensitive to such variations. Listed below are the energy releases which would occur for various selected flooding times:

Flooding Time (Seconds)	Average Reactivity Rate (Per Second)	Energy Release (Megajoules)
1.0	8%	290
0.8	10%	315
0.6	13.3%	350
0.5	16%	380
0.4	20%	420
0.25	32%	570

Because this variation in energy is small, it is believed that the value of 400 megajoules is a conservative estimate and this value is used to calculate the results to be expected from such a "maximum credible accident".

C. EXCURSION STUDY

The method of calculating the effects of an excursion is explained in Appendix A. It is shown that the moderator remains virtually unheated in an excursion, and that the fuel element temperature rise has a negligible effect on any excursion. Therefore, in an uncontrolled excursion, shutdown will not occur until the fuel elements have been partially melted or vaporized. The important shutdown mechanisms are found to be:

Removal of molten fuel element material by gravity.

Removal of fuel element material by the pressure exerted by the vaporized metal.

Displacement of the moderator and lead reflector by the pressure exerted by the vaporized metal.

The results of these calculations are shown in Figs. 21 and 22, in which the excursion energy is plotted against reactivity and reactor period.

The calculations were made for the unflooded core. For a given value of reactivity, the presence of water in the coolant channels will always decrease the magnitude of the excursion. Formation of steam, occurring before fuel element melting, will allow moderator displacement (in the form of ejecting water out the tube ends) to occur at a low energy level. Filling the voids with water will also increase the pressure tending to disassemble the reactor after metal vaporization begins.

At excursion energies of less than 100 magajoules some of the fuel element material will be melted. The core structure will remain intact (though perhaps with small leaks) and nearly all the fission products will be contained within the gas loop. At excursion energies much above 100 megajoules, however, the reactor will be shut down by the explosive ejection of fuel out of the ends of the pressure tubes. Complete rupture of the core structure may be expected. Most of the fission products that escape the molten fuel elements will be initially introduced into the pool. There, some will condense, but the more volatile products will bubble to the surface and escape into the reactor building. The building will remain intact, but these products may gradually leak out.

At a distance greater than 100 meters, the radiation dosage received from a cloud is due almost entirely to the fission products of steady state operation. The magnitude of the excursion must be known, however, in order to estimate the amount of fission product activity released from the fuel elements and subsequently the amount escaping from the gas loop, the pool and the reactor building.

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The maximum credible accident is considered to be the flooding accident, which adds reactivity at a rate of 16%/sec. As shown in Figure 20, this is equivalent to a step increase of 3.0%. With this energy release, nearly all the fuel element material is melted and about one-fourth is vaporized. The expected energy release is 380 megajoules. For the purposes of calculating the explosion damage and the release of activity, the maximum credible nuclear excursion is assumed to release 400 megajoules of energy.

The expected chemical energy release from steel-H_2O and aluminum-H_2O reactions is about 100 megajoules, as calculated in Appendix B. This energy release would be nearly simultaneous with the fission energy release. The potential energy from the evolved hydrogen is 275 megajoules. It is shown in Appendix B, however, that the hydrogen reaction would not occur as a single explosion, and would occur somewhat later than the excursion.

The total kinetic energy release is equivalent to the kinetic energy released by the explosion of 38 lbs of TNT. The explosion would crack the pool walls, but would not cause gross structural failure. The reactor building would be virtually undamaged, but large quantities of water, water vapor, hydrogen, and evolved gaseous fission products would be introduced into the building. With the exception of the noble gases, much of the fission product activity (particularly strontium and iodine) would remain in the molten fuel. Most of that which escaped the fuel would condense or otherwise remain in the water. Some might escape through the gas channels into the equipment room. For the purposes of calculating the distribution of radioactivity following the maximum credible excursion, it is pessimistically assumed that 10% of the total fission product activity is introduced into the atmosphere.

The maximum conceivable accident is one in which the near critical reactor is flooded in 0.17 seconds. No mechanism is known that would produce this condition, since it requires that all the coolant passages are simultaneously flooded with water under pressure, with a flow of 80 gallons per second. In this accident the fission energy release would be approximately 1100 megajoules, accompanied by a chemical energy release of 400 megajoules. The explosive damage to the reactor building would be severe. While the amount of fission product activity contained in the fuel (mostly from steady state operation) would not be much greater than that of the 400 megajoule excursion, a greater fraction of the activity would be introduced into the atmosphere.

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VI. DISTRIBUTION OF RADIOACTIVE PRODUCTS

A. WASTE PRODUCTS

1. Gaseous Waste

a. Disposal

In general, contaminated gaseous waste will be disposed of in the following manner:

(1) Contaminated gas leaks from the GCRE-I loop will be diluted in the facility exhaust ventilation and discharged from a 150-ft stack located adjacent to the reactor building.

(2) Contaminated gas withdrawn from the loop will be stored in a holdup tank. This gas will be analyzed for radioactivity level and for identification of emitter. It will then be discharged from a special 150-ft stack located 200 ft from the reactor building. The discharge will be at a safe rate, determined by climatic conditions and by the nature and level of the activity in the gas.

b. Effluents Expected in Normal Operation

During normal operation, corrosion will result in the introduction of particulate matter from the reactor into the gas stream. The concentration of this material must be held at a low level for two reasons: first, the activity in the external loop must be held as low as possible for ease of maintenance; and second, material in the gas stream will cause erosion, multiplying the effect. A filter on a bypass stream will maintain the concentration at a low level.

The amount of argon 41 activity is dependent on the purity of the nitrogen coolant. If the argon content is held below 0.1% the total argon activity will not exceed 0.06 curies. Carbon 14 will be created in the nitrogen coolant by the (n,p) reaction. However, in 2 years of reactor operation, only 0.15 curies will have been formed. These estimates do not consider fission product diffusion (particularly iodine and xenon) through the stainless steel. It is difficult to obtain any experimental data concerning these diffusion rates. On the basis of calculated values this additional contamination will not be a serious problem.

A controlled coolant bleed from the loop, used in conjunction with the particulate filters, will maintain the radioactivity level of the loop gas at less than 10 curies. The integrated activity of the gas bled from the loop may run as high as 80 curies per day. This gas will be collected in the gas holdup tank before disposal from the stack.

c. Effluents with 10^{-5} Leak

After equilibrium shutdown and several days decay, the total activity in the reactor will be about 2 megacuries. A fuel element leak in which 10^{-5} of the total fuel element activity was introduced into the gas stream would correspond to 20 curies (after several days decay). The cleanup

procedure will consist of washing the reactor and loop with water to remove corrosion products. If the external loop activity is not sufficiently lowered by this procedure, the valve between the loop and reactor will be closed and the loop washed with a dilute acid. The effluents from these procedures will be chemically treated. This procedure is designed for the routine disposal of activity levels to 10 curies, so that little difficulty should be involved from this standpoint.

d. Effluents with 10^{-2} Leak

A fuel element leak of 10^{-2} would be a major catastrophe. It could occur as the result of a partial fuel-element meltdown, due to a reactor excursion or a control system failure. After several days decay the activity released would be 20 kilocuries. Since it is difficult to dispose of quantities of activity greater than 10 curies, a leak of this magnitude might require 3 or 4 months for decontamination. At least 10% of this activity would be disposed of as gaseous effluents to the stack over a period of several weeks, providing that the activity had not escaped during the accident.

e. Stack Height Effects

In normal reactor operation, a small amount of radioactive material will be discharged through the stack. The stack height is sufficient to ensure that these products are carried away from the site and diluted by the air, rather than being deposited near the site by eddies. In general, a downwash can be prevented if the stack is at least 2½ times the height of any structure in the immediate vicinity (AECU 3066). This sets a minimum height of approximately 100 ft.

Increasing the stack height above 100 ft has a significant effect on the direct radiation received at points near the stack. The down-wind radiation due to washout or fallout of the cloud is nearly independent of stack height. The cloud concentration is also nearly independent of stack height so that the inhalation dose received by a person in this cloud is also nearly independent. However, additional stack height will hold the cloud above the ground until it has diffused into a larger volume, or lower concentration.

In order to obtain lower radiation levels than would exist with a minimum 100 ft stack, a 150 ft stack is provided. The velocity of discharge is about 50 ft/sec, so that under normal wind conditions the effective stack height is several hundred feet higher.

2. Liquid Waste

The liquid waste will consist of contaminated water from washing the loop and reactor before working on the loop, before opening the reactor to the pool, or from decontamination operations. Disposal will be made in the following manner:

a. Contaminated water will be transferred from the reactor loop system to a holdup tank.
b. The contamination level in the water will be determined.
c. An isotopic determination will be made of the principal contaminants.

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d. Steps will be taken to reduce the contamination level to a value sufficiently low to permit dumping the water. Chemical methods will be used as determined by the nature of the contaminants and by the contamination level.

Deionized water will be used in the reactor pool to reduce the activation of mineral ions.

3. Solid Waste

Solid waste will be composed of such items as loop-gas particulate filters, resin columns, counting discs, and materials used to clean spills. Solid-waste disposal procedures will be:

a. Wherever practical, dry radioactive waste will be wrapped or bagged.
b. Dry-waste parcels will be collected in receptacles that can be buried.
c. When a number of waste receptacles have accumulated they will be hauled away for burial under the supervision of the Site Survey group of the Health and Safety Division, Idaho Operations Office.

4. Spent Fuel Elements

Spent fuel elements will be stored in racks located at the bottom of the reactor pool. Fuel elements will be removed from the pool by lowering a shielded shipping cask into the pool to receive the elements.

B. PRODUCTS FROM MAXIMUM ACCIDENT

1. Assumptions

The following assumptions were made in calculating the radiation effects that result from a maximum credible accident:

a. The reactor has been operating at 2 megawatts for 10^4 hr prior to the excursion. Most fission products have reached equilibrium values.
b. The integrated excursion energy is 400 megajoules.
c. Ten percent of the total fission product activity escapes from the reactor vessel, up through the pool into the reactor room. This activity subsequently is released into the atmosphere as a cloud having a 20 meter radius.
d. The diffusion theory of O. G. Sutton is taken to be valid.
e. Wind shear is neglected.

Assumptions, a, b, and c represent maximum conditions. Hence the effects predicted by the following calculations are believed to be pessimistic, and represent maximum (rather than probable) effects from a maximum credible accident. The methods used in calculating the effects are those shown in Appendix C.

356 060

2. Radiation Levels at Various Downwind Distances

The activity laden cloud that results from the accident drifts downwind from its origin. Three radiation mechanisms were investigated to determine the radiation that will be received by a person at various distances downwind from the explosion: direct gamma radiation from the cloud; radiation due to washout from the cloud; and inhalation of particular isotopes. In all cases, the values shown are those that are maximum for the given distance (the point directly downwind from the explosion). No shielding is postulated other than that of the air. The radiation is mostly that which results from the steady-state fission products.

a. Direct Gamma Radiation from Cloud

Calculations were made by employing the nomographs that were constructed by J. Z. Holland of the U. S. Weather Bureau (Reference AECU-3066). The total integrated radiation was determined for the different downwind distances by the use of the nomographs. Wind velocity was varied between 1 meter/sec and 20 meters/sec. Sutton stability values of 0.2 and 0.5 were used, corresponding to a large lapse condition and a large inversion, respectively. The results are shown in Figure C-1, Appendix C.

Under the worst conditions, the maximum dosage is found to be approximately 12 roentgens at 200 meters, 6 roentgens at one mile, and 1 roentgen at ten miles.

b. Radiation Due to Fallout and Rainout from the Cloud

It was assumed in calculating the radiation that results from washout that, as the radioactive cloud passes over the observer, a given fraction of the radioactive material is deposited per unit distance of cloud travel. The equations that are used (Reference KAPL-1045) overestimate the concentration at all distances. The cloud radii used are obtained from Sutton's diffusion equations. The resulting dose rates are shown in Figure C-2, Appendix C.

The worst condition is shown to be a rainout with a temperature inversion and high winds, a virtually impossible condition. The worst probable rain-out condition would cause an initial dose rate of about 400 R/hr at 200 meters, 10 R/hr at 1 mile and less than 0.5 R/hr at 10 miles.

c. Inhalation of Particular Isotopes

Calculations were made of the inhalation dose received due to six important long-lived fission product isotopes containing about 85% of the inhalation activity. These calculations were made for a wind velocity of 2 meters/sec. The results are shown as a function of the distance downwind in Figures C-3, C-4, C-5 and C-6, Appendix C. At one mile, the infinite time inhalation doses are found to be:

SRS 061

		Lapse	Inversion
Sr-90 and Y-90	(half life = 28 years)	10 R	700 R
Ce-144 and Pr-144	(half life = 280 days)	9 R	500 R
I-131	(half life = 8 days)	5 R	300 R

With the worst weather conditions, a lethal inhalation dose could be received at distances out to about one mile. For average conditions, this lethal distance would extend out to several hundred meters.

An above tolerance I-131 inhalation dose could be received at distances out to 2.5 miles for a lapse condition. It could occur out to 50 miles for a large inversion, if the inversion were continuous for this distance during a period of many hours. This condition is extremely unlikely.

3. Radiation Received by Reactor Personnel

An unshielded person who is within several hundred feet of the reactor would probably receive a lethal radiation dose due to the combined effects of direct radiation, particle fallout, and inhalation.

Following an excursion, the direct gamma radiation received by personnel in the control room next to the reactor room wall would amount to about 113 roentgens within ten (10) minutes. An additional amount would be received due to inhaling of gaseous fission products (krypton, xenon) which are not removed by the filter. Evacuating these personnel to the counting room would decrease this dose to approximately 18 roentgens in a 10-minute period.

4. Conclusions

The dose calculations for the maximum credible accident are based on assumptions of long term reactor operation and the escape of 10% of the fission products. Therefore, the predicted effects represent maximum (rather than probable) results. On this basis, severe or lethal radiation injuries could occur among the reactor personnel, particularly if they are unshielded. Immediate evacuation of the affected downwind areas would be required if the accident occurred during a temperature inversion, with proper wind direction. In most cases, no serious radiation injuries would be expected outside of the reactor site, although a lethal inhalation dose could be received out to about one mile from the reactor if all the "worst" conditions occurred simultaneously.

356 062

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LETTER	DATE	CHANGE	BY	CHK'D

ANP AREA
IET
FET
AMA
POLE LINE ROAD
LPTF
HIGHWAY
UTAH POWER & LIGHT Co 44 KV
IDAHO CENTRAL
HOWE
132 KV. LINE
LINCOLN BLVD.
132 KV. LINE
NRF
ARCO
44 KV. LINE
AT&T Co TELEPHONE LINE
U.S. 20
MTR-ETR
CPP
AREA
TREAT
EBR II
ARBOR
TO IDAHO FALLS
CF
12.5 KV.
132 KV.
GCREI
ALPR
BORAX
EBR I
138 KV. LINE
U.S. 26
ATOMIC CITY
N

0 1 2 3 4 5

COPY

356 063

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FIG.— 1

No. Reqd.	Next Assy.	Model	
Heat Treat	Finish		
Tolerance Unless Otherwise Noted			
.xx ± .05			
Linear .xxx ± .010			
Tol. Angular ±			
▽ Indicates Surface Roughness			

MATERIAL

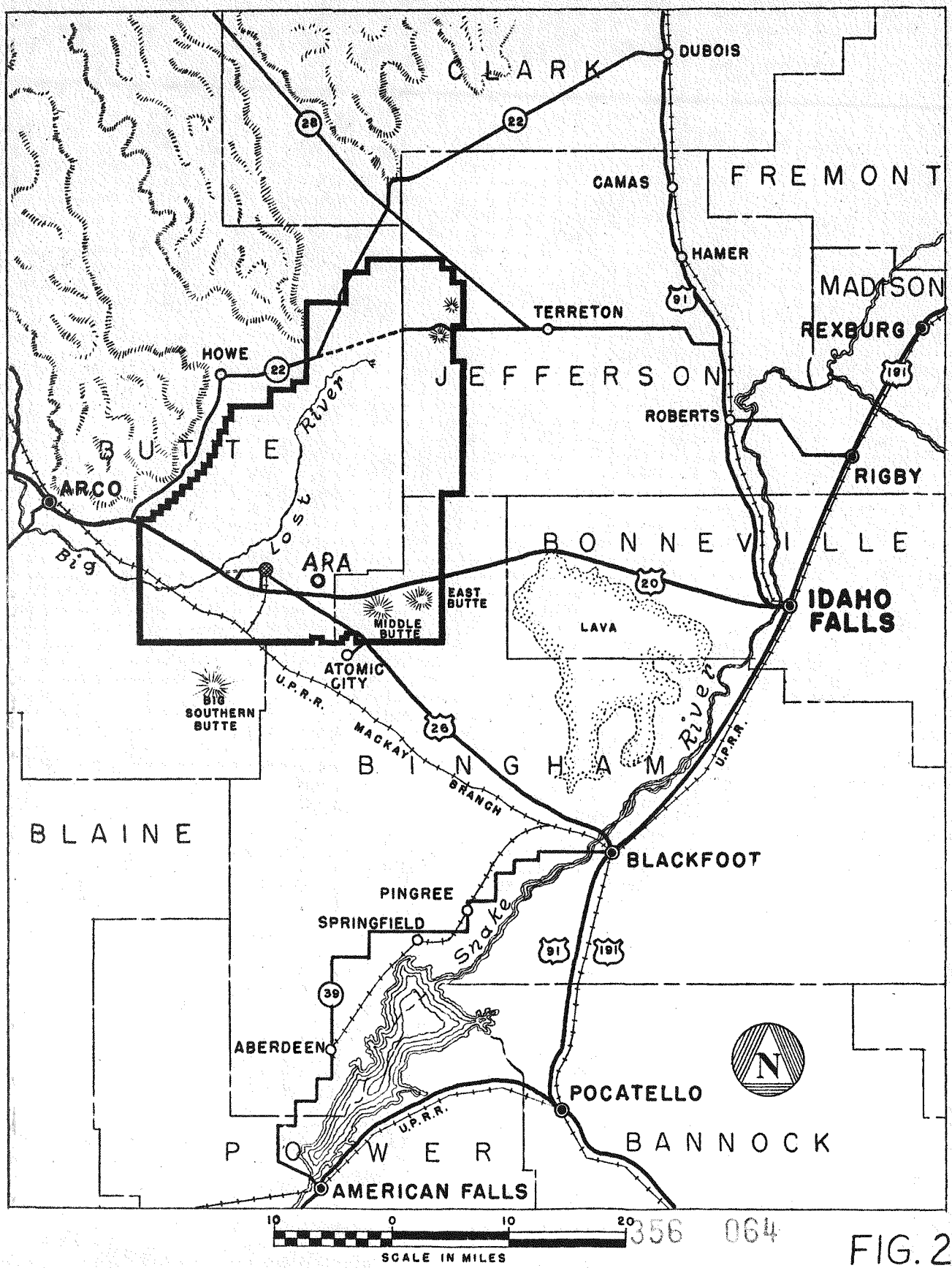
NRTS
IDAHO

AEROJET-GENERAL
NUCLEONICS
SAN RAMON
CALIFORNIA

Date 6-12-59
Drawn CDA
Approved
Dwg. Size F
880-AREA-501
Part Number

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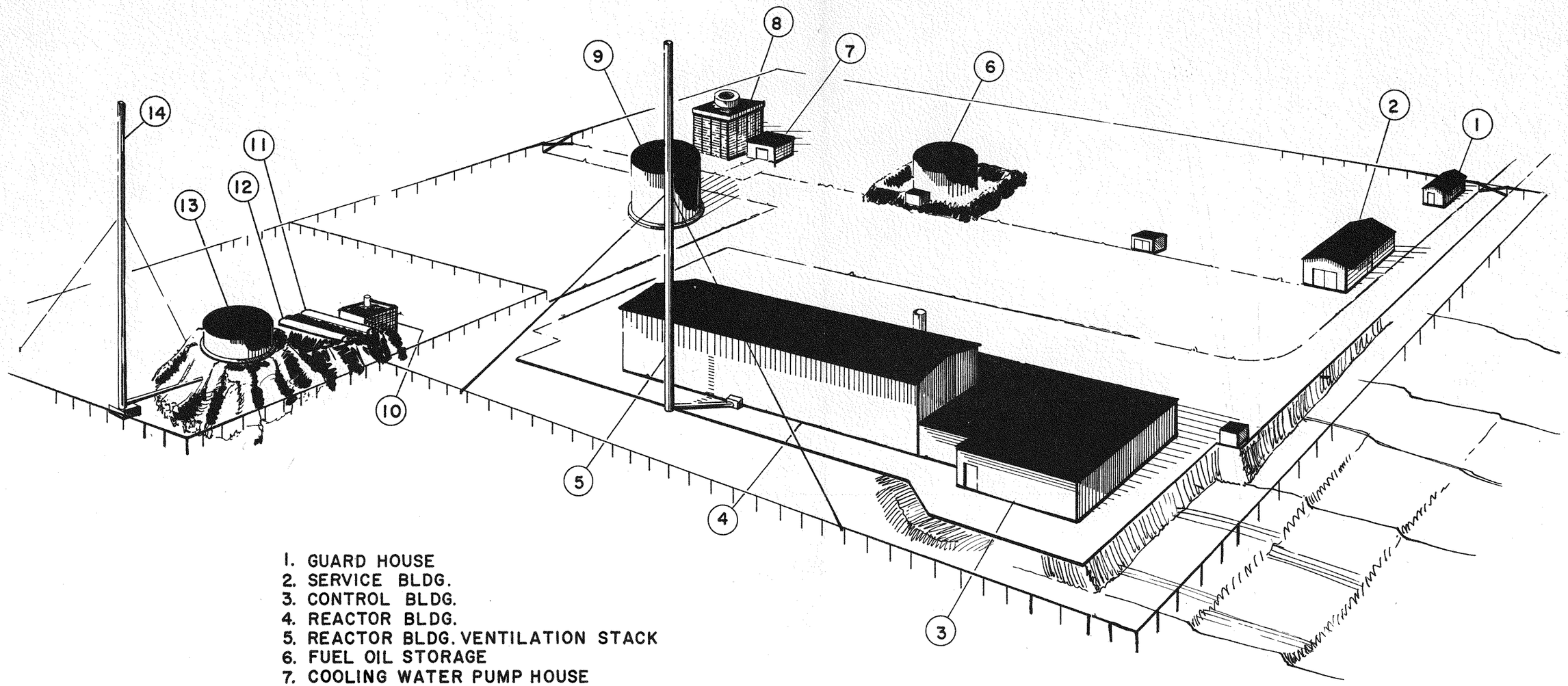


356 064

FIG. 2

U. S. ATOMIC ENERGY COMMISSION IDAHO OPERATIONS OFFICE IDAHO FALLS, IDAHO	TITLE NATIONAL REACTOR TESTING STATION VICINITY MAP
APPR. [signature] DIRECTOR DIV. OF ENG'R & CONST.	APPR. [signature] CHIEF ENGINEERING BRANCH
DRAWN L.N.P. CH'CK'D R.J.P.M. APP'VD	DATE OF ISSUE REDRAWN DECEMBER 10, 1956 SCALE AS SHOWN SHEET 1 OF 1 DWG. NO. B-60-102-IDO-2

LETTER	DATE	CHANGE	BY	CHD	



1. GUARD HOUSE
2. SERVICE BLDG.
3. CONTROL BLDG.
4. REACTOR BLDG.
5. REACTOR BLDG. VENTILATION STACK
6. FUEL OIL STORAGE
7. COOLING WATER PUMP HOUSE
8. COOLING TOWER
9. UTILITY WATER STORAGE
10. RADIOACTIVE WASTE PUMP HOUSE
11-12. LIQUID RADIOACTIVE WASTE TANK
13. WASTE GAS HOLDER
14. WASTE GAS STACK

GCRE-1 AREA PERSPECTIVE

FIG.-3

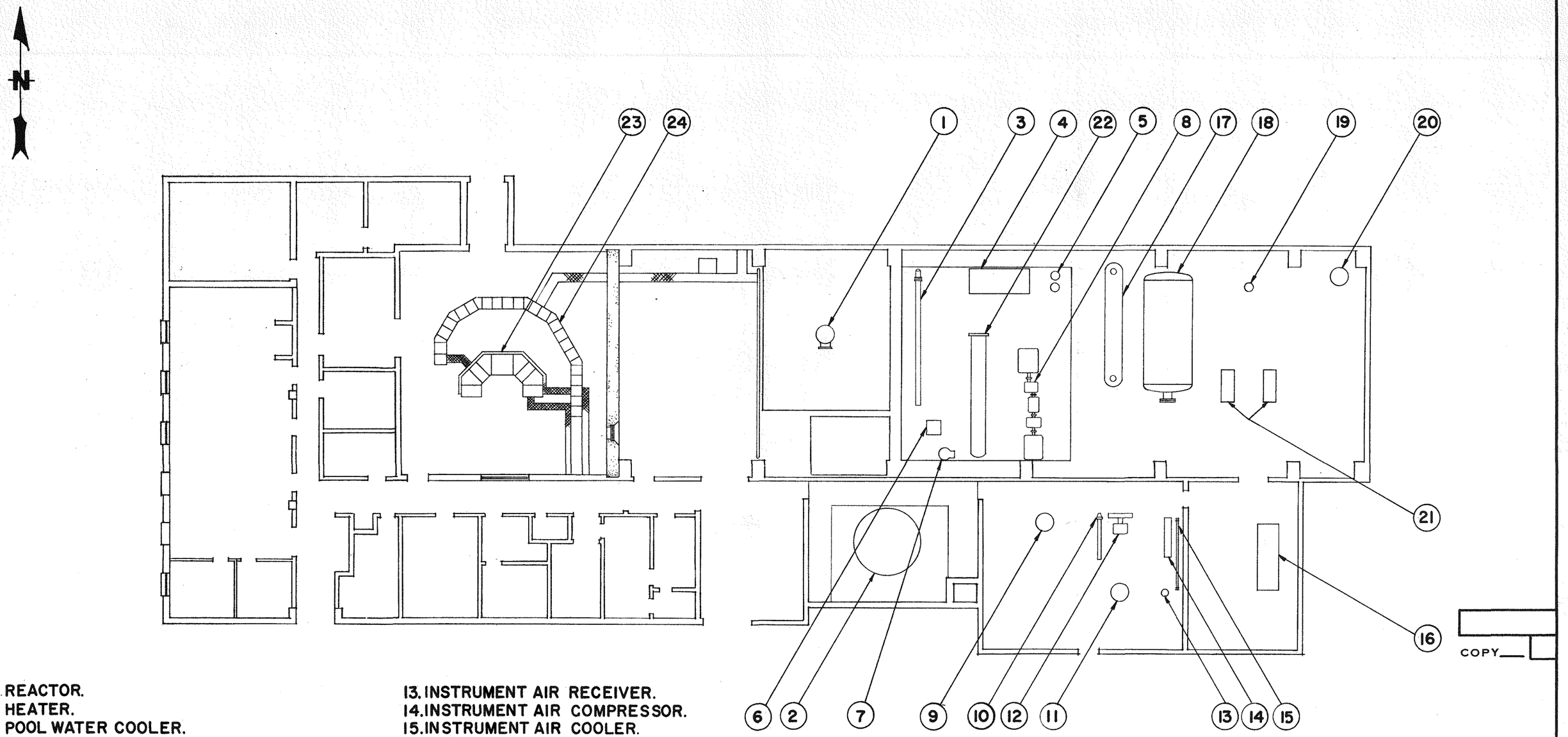
356 065

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LETTER	DATE	CHANGE	BY	CH'D	



1. REACTOR.
2. HEATER.
3. POOL WATER COOLER.
4. GAS DRYER.
5. GAS FILTERS.
6. SHUTDOWN COOLING BLOWER.
7. SHUTDOWN COOLING SYSTEM COOLER.
8. MAIN COMPRESSOR & DRIVE.
9. CLEANING SOLUTION MAKE-UP TANK.
10. UTILITY AIR COOLER.
11. UTILITY AIR RECEIVER.
12. UTILITY AIR COMPRESSOR.
13. INSTRUMENT AIR RECEIVER.
14. INSTRUMENT AIR COMPRESSOR.
15. INSTRUMENT AIR COOLER.
16. EMERGENCY DIESEL GENERATOR.
17. WATER-TO-WATER HEAT EXCHANGER.
18. DEMINERALIZED WATER SURGE DRUM.
19. RAW WATER DEMINERALIZER.
20. POOL WATER DEMINERALIZER.
21. DEMINERALIZED WATER CIRCULATING PUMP.
22. GAS-TO-WATER HEAT EXCHANGER.
23. CONSOLE.
24. INSTRUMENT RACKS.

COPY

GCRE-1
REACTOR AND CONTROL
BLDG. FLOOR PLAN
FIG.-4

356 066

			DRAFTSMAN	DATE
No. Reqd.	Next Assy.	Model	CHECKED	
Heat Treat		Finish	STRESS	
Tolerance unless otherwise noted			ENGINEER	
Linear Tol.	.XX	± .03	APPROVED	
	.XXX	± .010	PRODUCTION	
Angular Tol.			MATERIALS	

Indicates surface roughness. Finish per MIL-STD-10 unless otherwise noted

MATERIAL: UNCLASSIFIED

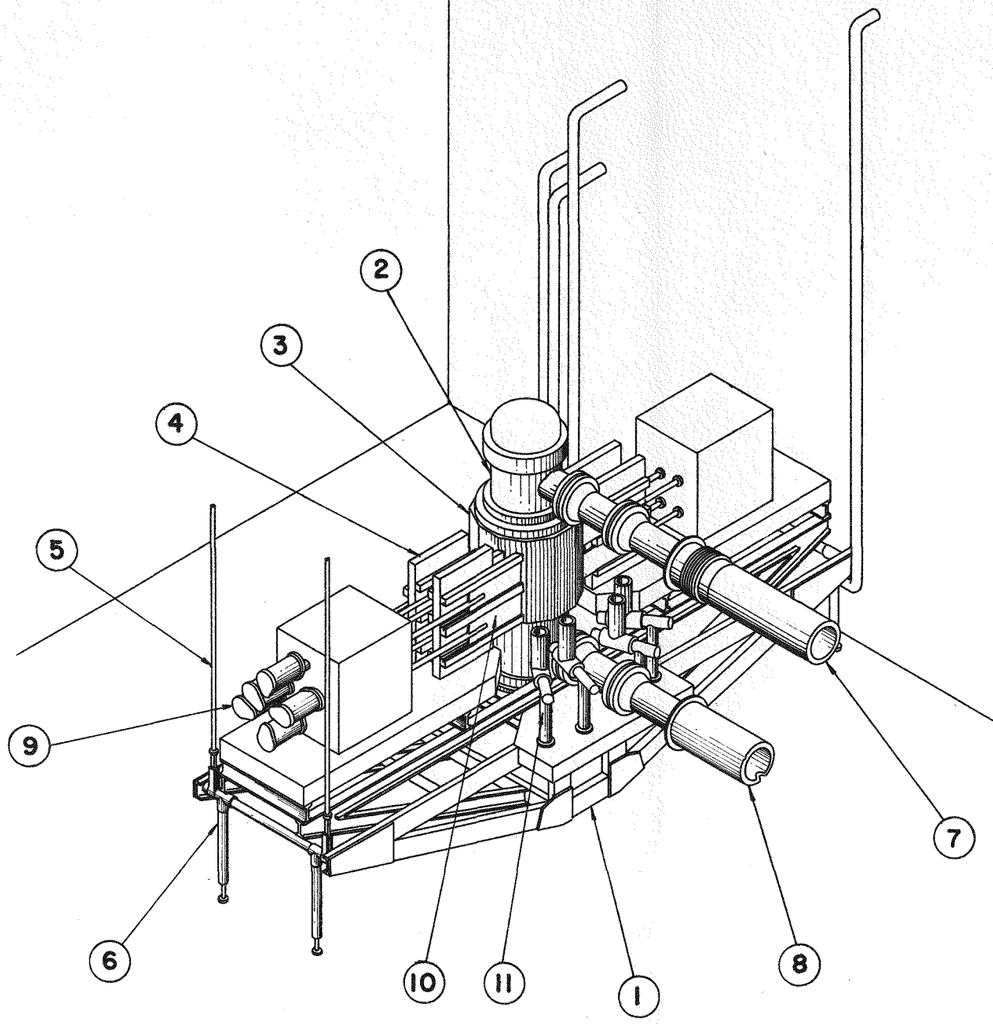
AEROJET-GENERAL NUCLEONICS
SAN RAMON CALIFORNIA

Similar to | Scale | Dwg. Size: D | Release Date | Est. Weight | Drawing Number

Break all sharp edges

UNCLASSIFIED

LETTER	DATE	CHANGE	BY	CH'D	



1. REACTOR SUPPORT STAND.
2. REACTOR (INLET PLENUM).
3. LEAD SHIELD.
4. GUIDE ASSEMBLY.
5. STAND SUPPORT RODS.
6. STAND SAFETY SUPPORTS.
7. INLET COOLANT GAS DUCT.
8. OUTLET COOLANT GAS DUCT.
9. ROD ACTUATOR (TYP).
10. ROD ABSORBER BLADE (TYP).
11. NEUTRON DETECTOR (TYP).

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356 067

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NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

No. Reqd.	Next Assy.	Model	DRAFTSMAN	DATE
			CHECKED	
HEAT TREAT	FINISH		STRESS	
TOLERANCE UNLESS OTHERWISE NOTED			ENGINEER	
LINEAR TOL.	.XX	± .03	APPROVED	
	.XXX	± .010	PRODUCTION	
ANGULAR TOL.			MATERIAL	
INDICATES SURFACE ROUGHNESS FINISH PER MIL-STD-10 UNLESS OTHERWISE NOTED				

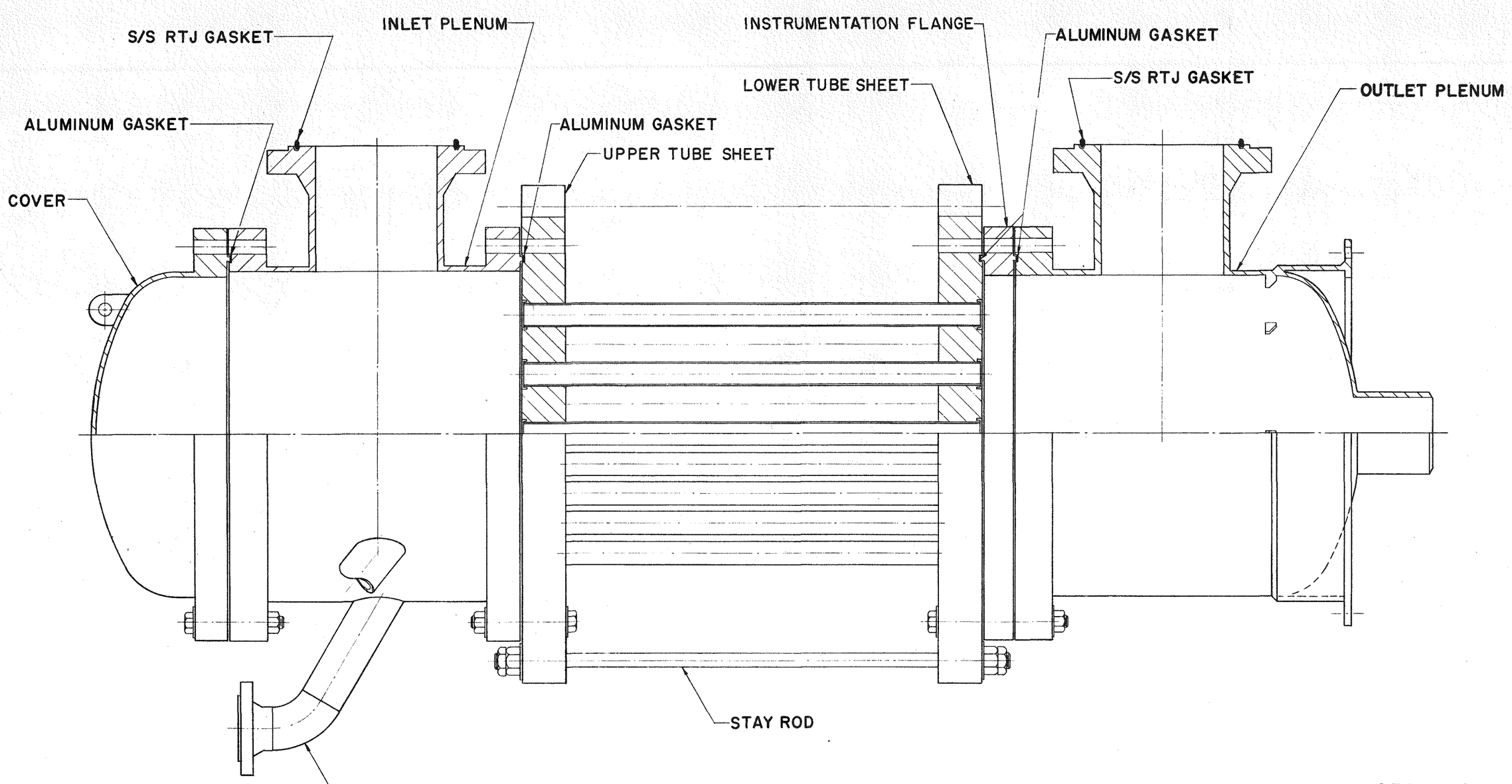
MATERIAL

G.C.R.E. I REACTOR POOL PERSPECTIVE FIG.-5

AEROJET-GENERAL NUCLEONICS
SAN RAMON CALIFORNIA

SIMILAR TO	SCALE	DWG. SIZE	DRAWING NUMBER
RELEASE DATE	CAL. WEIGHT	D	
	NET WEIGHT		

LETTER	DATE	CHANGE	BY	CH'D	



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REACTOR VESSEL ASSEMBLY
HALF SECTION

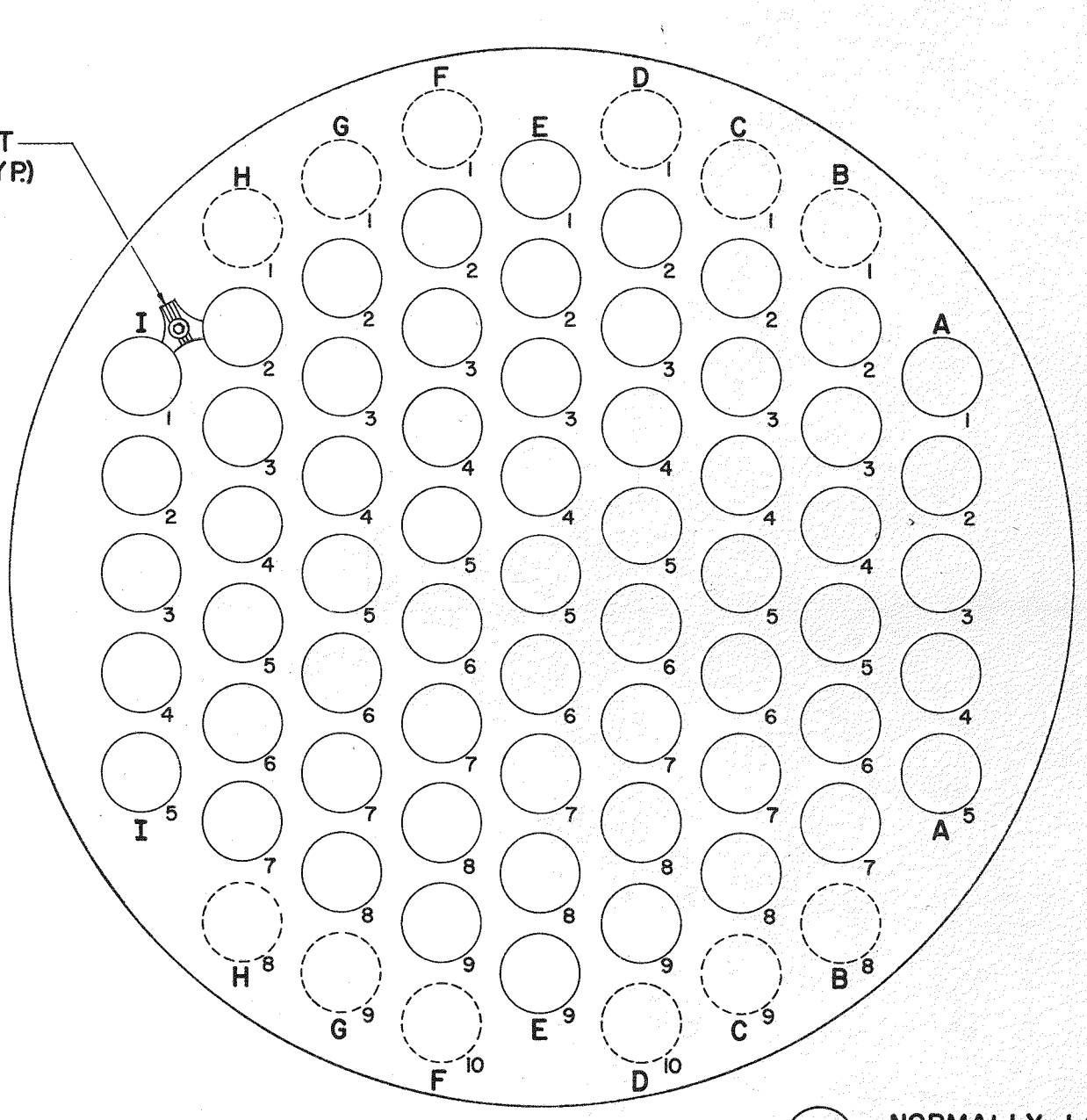
NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

No. Reqd.	Next Assy.	Model	J. JAGGER DRAFTSMAN	Date	MATERIAL
Heat Treat	Finish		Checked		GCRE-1 VESSEL FIG.6
Tolerance unless otherwise noted			Stress		AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA
Linear .XX	± .03		Engineer		
Tol. .XXX	± .010		Approved		
Angular Tol.			Production		Similar to / Scale / Dwg Size D
√ Indicates surface roughness 63 finish per MIL-STD-10 unless otherwise noted			Material		Release Date / Cal. Weight / Act. Weight / Drawing Number

RELEASED
DECLASSIFIED

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

GCRE CORE LAYOUT

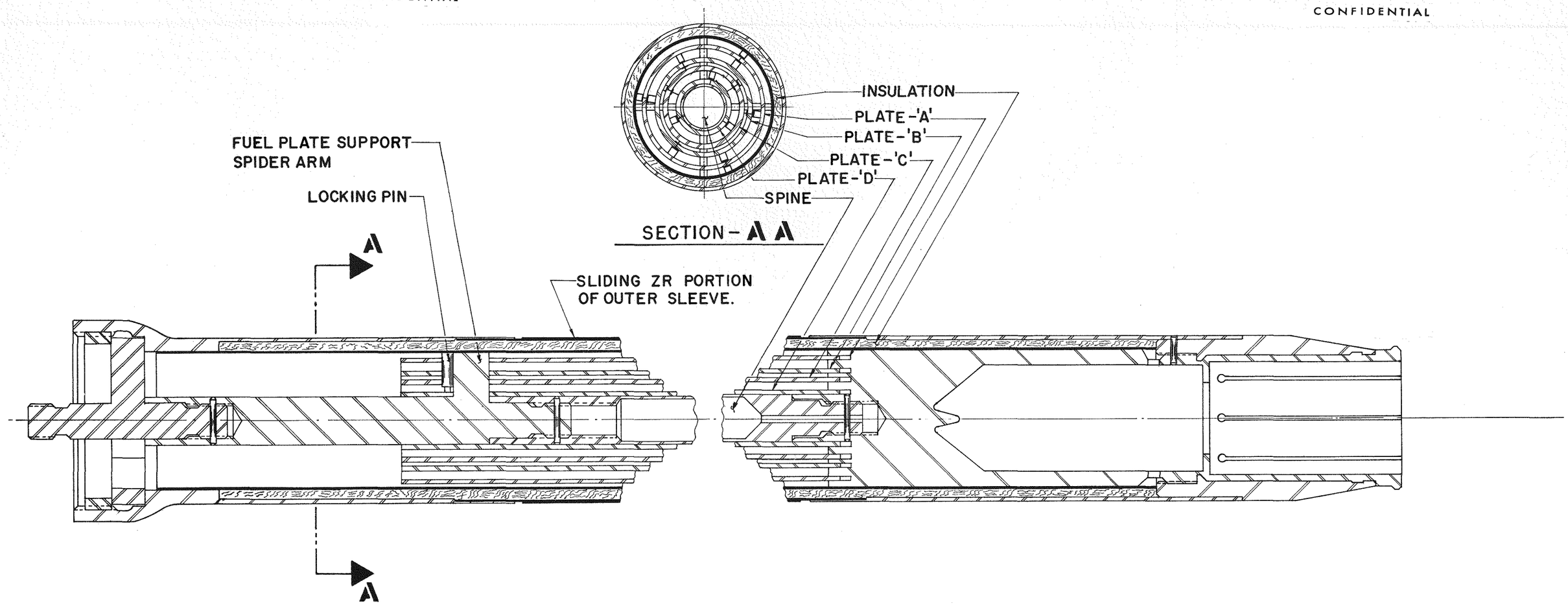


GCRE-I CORE LAYOUT

FIG.-7

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GCRE-I
REFERENCE FUEL ELEMENT
FIG.-8

356 070

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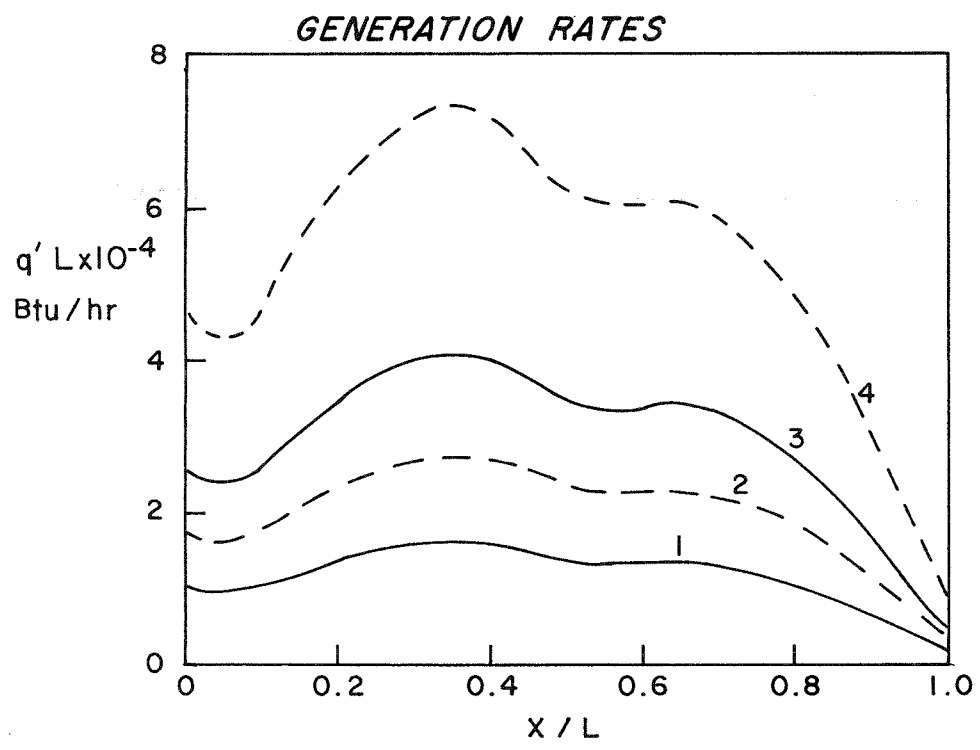
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359 071

STEADY STATE GCRE TEMPERATURES
IBM 650 CALCULATIONS
RUN NO. 265
FULL POWER,
FULL FLOW PROBLEM

CONFIDENTIAL

TYPE ELEMENT -- MFG-1
LOADED LENGTH -- 28 inches
TOTAL REACTOR POWER (not including radiation) 2.08 Mw
NUMBER OF ELEMENTS -- 61
ELEMENT POWER (average generating element) 1.164×10^5 Btu/hr
INLET GAS TEMPERATURE -- 800°F
GAS MIXED MEAN OUTLET TEMPERATURE -- 1205°F
MODERATOR TEMPERATURE -- 130°F
TOTAL REACTOR FLOW RATE -- 6.32×10^4 lb/hr
TOTAL ELEMENT FLOW RATE -- 1037 lb/hr
PRESSURE DROP -- 874 lb/ft^2
TOTAL LENGTH -- 31 inches



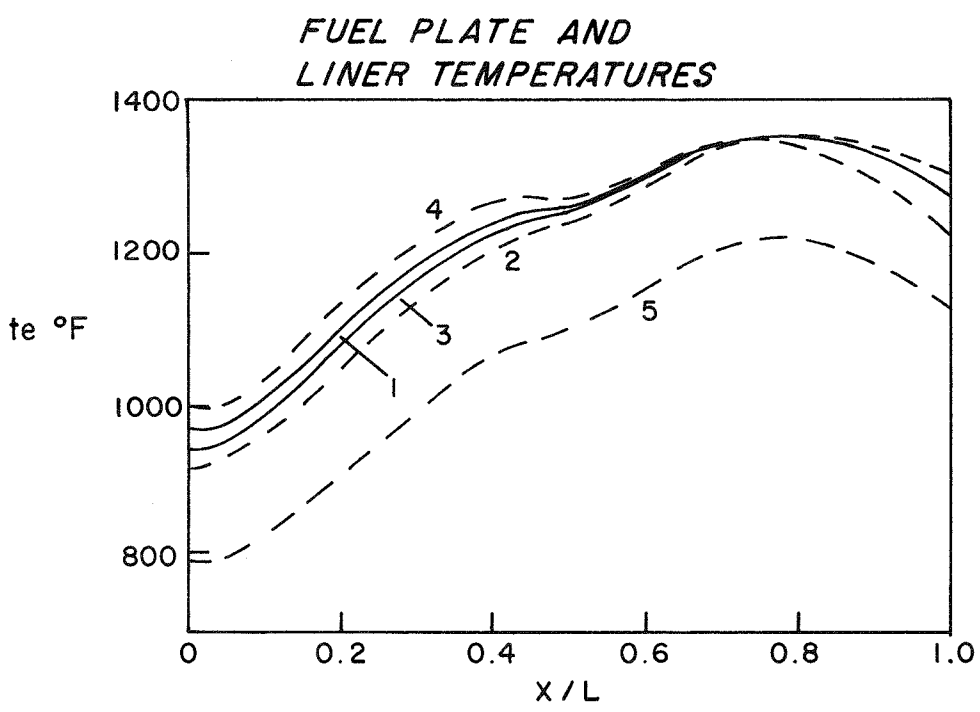
INDIVIDUAL CHANNEL CONDITIONS

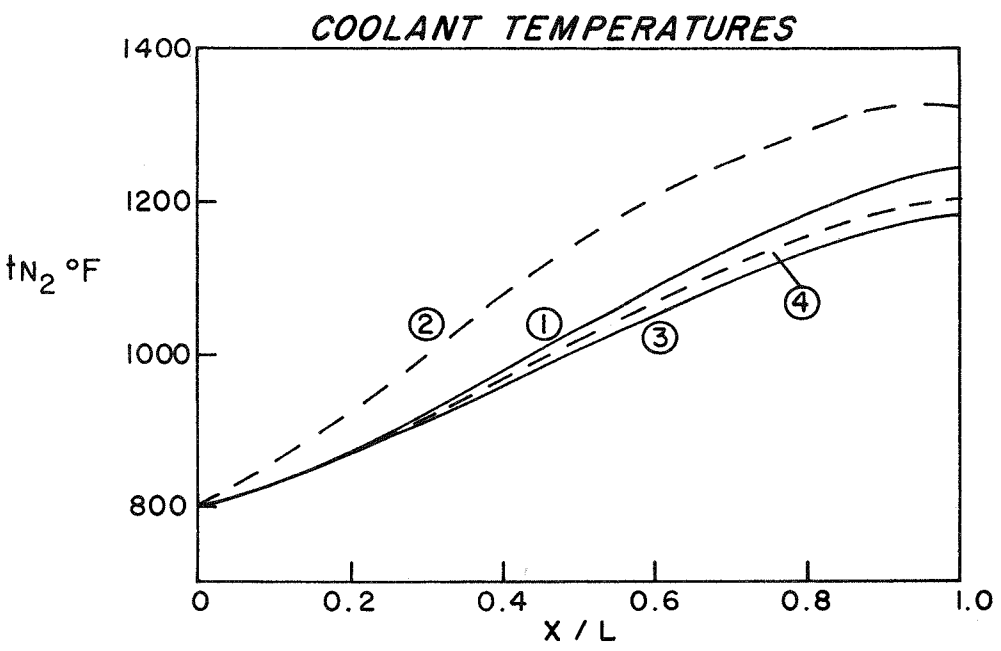
FLOW RATES, FLOW AREAS, EQUIV. DIAMETERS

location	w lb/hr	A_c ft^2	D_e ft
①	216	1.073×10^{-3}	1.052×10^{-2}
②	114	0.754×10^{-3}	0.660×10^{-2}
③	469	2.003×10^{-3}	1.361×10^{-2}
④	238	1.396×10^{-3}	0.802×10^{-2}

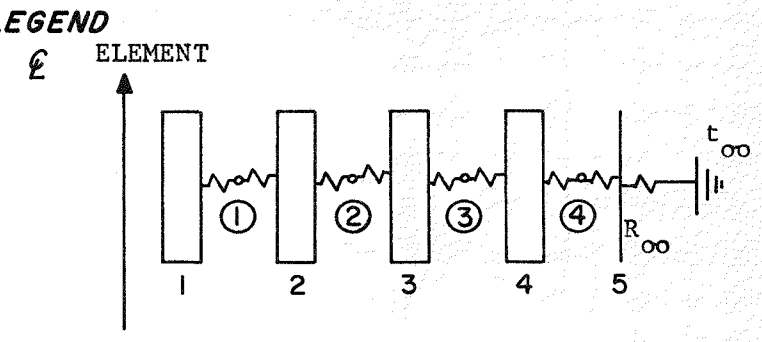
HEAT TRANSFER COEFFICIENTS, HEAT TRANSFER AREAS

location	h(convection) Btu/hr ft^2 $^\circ$F	A_h ft^2
1 ①	204	0.289
2 ①	198	0.398
2 ②	176	0.453
3 ②	169	0.508
3 ③	215	0.512
4 ③	140	0.632
4 ④	14.3	0.720
$1/R_{\infty\infty}$	2.7 Btu/hr $^\circ$F	





LEGEND



$R_{\infty\infty}$ = 1/UA TOTAL HEAT TRANSFER RESISTANCE BETWEEN THE INNER LINER AND THE MODERATOR -- hr ft^2/Btu
$t_{\infty\infty}$ = MODERATOR TEMPERATURE -- $^\circ$F
q' = HEAT GENERATION RATE PER UNIT LENGTH -- Btu/hr ft
L = LENGTH OF FUEL ELEMENT MEAT -- ft
X = DISTANCE ALONG FUEL ELEMENT FROM THE INLET -- ft

NOTES

POWER DISTRIBUTION IN AN ELEMENT FROM THE GCRE-I CRITICAL

AXIAL POWER DISTRIBUTION FROM THE GCRE-I CRITICAL

GAP SPACING (@ operating temperature)

Location	width - inches
1	0.089
2	0.045
3	0.098
4	0.052

Fig. 9

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CONTROL ROD WITHDRAWAL SEQUENCE

STEP	ROD	DISTANCE	WORTH % ΔK/K	TIME	MAX. % ΔK/K PER MIN.	K eff * AT END OF STEP
1	GANG I	FULL OUT	8.0	15 MIN.	1.12	.875 - .895
2	GANG II	3.75" OUT	5.0	3.75 MIN.	1.32	.925 - .945
3	GANG II	6.5" OUT	2.5	2.75 MIN.	.91	.950 - .970
4	#3 SD	FULL OUT	.625	8.5 MIN.	.07	.956 - .976
5	#4 SD	FULL OUT	.625	8.5 MIN	.07	.963 - .983
6	#5 SD	FULL OUT	.625	8.5 MIN.	.07	.969 - .989
7	#6 SD	FULL OUT	.625	8.5 MIN.	.07	.975 - .995
8	SB	FULL OUT	.3	2. MIN.	.17	.978 - .998
(1) 9	FINE	1"/min to 5.5" out	.2		.04	.98 - 1.0
10	#1 S	FULL OUT	1.0	23 MIN.	.15	.99
11	#2 S	FULL OUT	1.0	23 MIN.	.15	1.0

* RANGE OF VALUES COVERS LIMITS OF Keff AT CRITICAL FROM .02% (ALL RODS OUT EXCEPT ½ F) TO 2.2 % (ALL RODS OUT EXCEPT 2S AND ½ F).

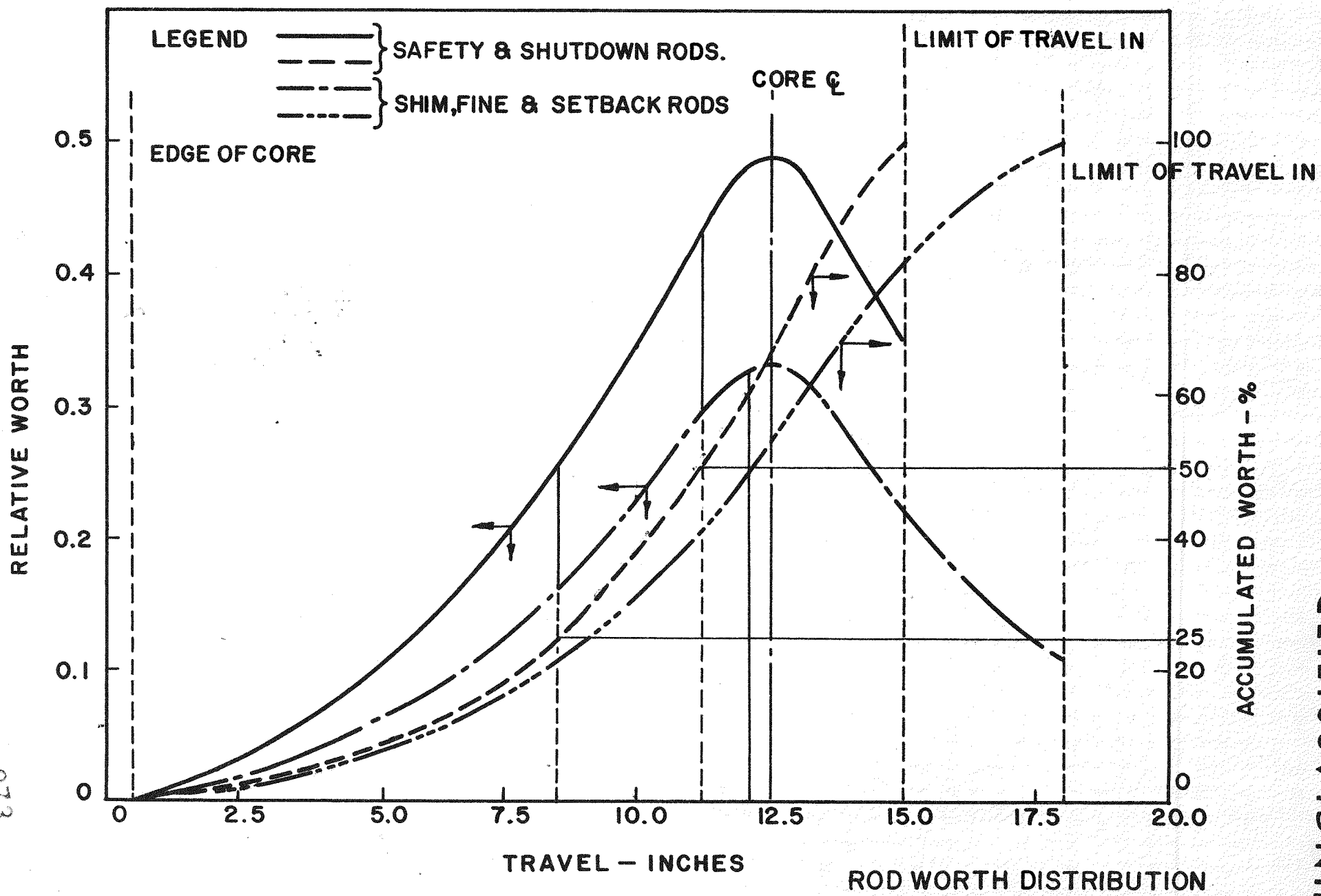
(1) TIMED WITHDRAWAL BEGINS AT THIS POINT.

FIG.-10

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356 072



ROD WORTH DISTRIBUTION

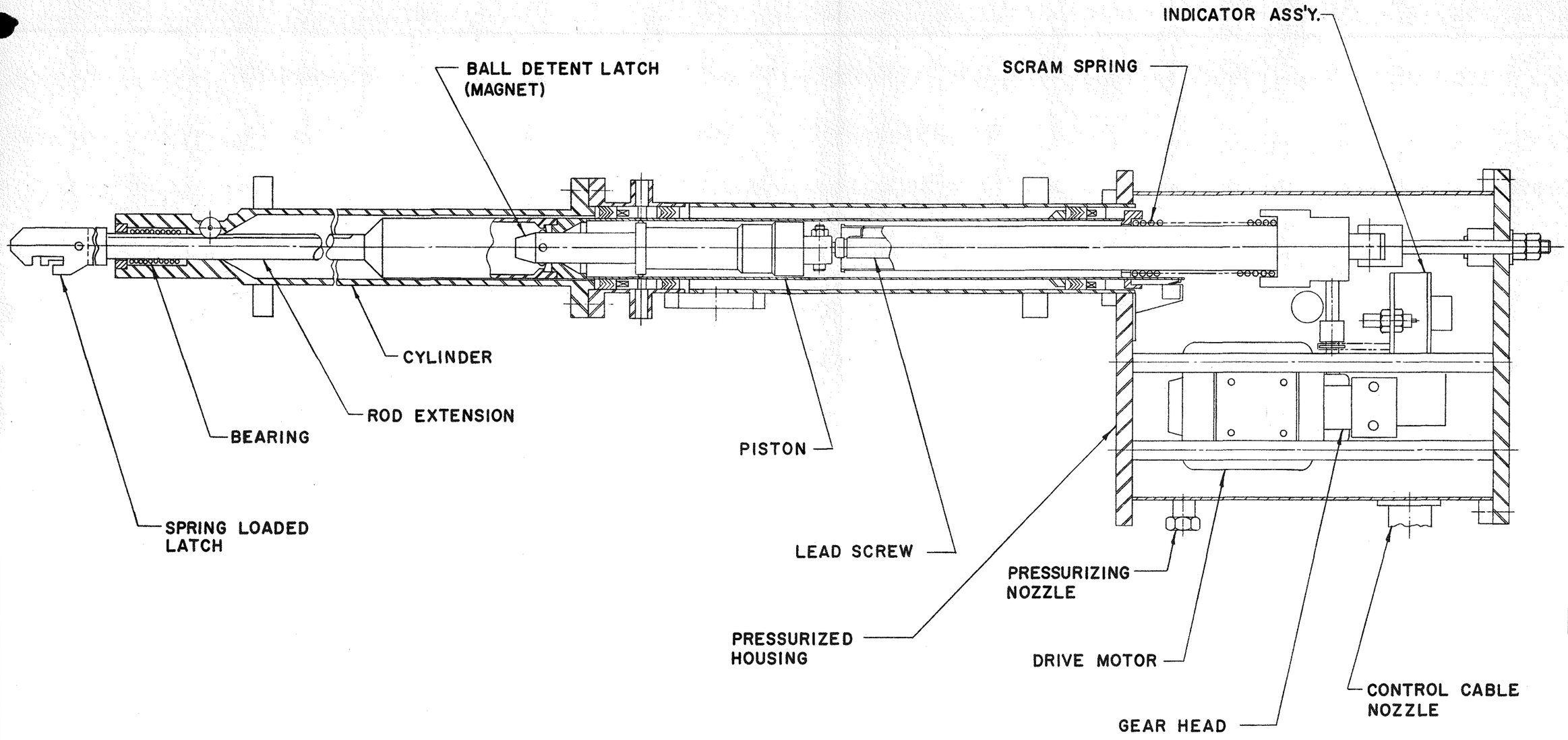
FIG-11

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ORNL-LR-DWG 356 013

LETTER	DATE	CHANGE	BY	CH'D



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NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

No. Reqd.	Next Assy.	Model	DRAFTSMAN	DATE
			CHECKED	
Heat Treat	Finish		STRESS	
TOLERANCE UNLESS OTHERWISE NOTED			ENGINEER	
LINEAR TOL.	.XX	± .03	APPROVED	
	.XXX	± .010		
ANGULAR TOL.			PRODUCTION	
INDICATES SURFACE ROUGHNESS FINISH PER MIL-STD-10 UNLESS OTHERWISE NOTED			MATERIAL	

MATERIAL

SAFETY ROD ACTUATOR FIG. 12

AEROJET-GENERAL NUCLEONICS
SAN RAMON CALIFORNIA

SIMILAR TO	SCALE	DWG. SIZE	
RELEASE DATE	CAL. WEIGHT	D	DRAWING NUMBER
	ACT. WEIGHT		

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CONTROL ROD CHARACTERISTICS

		NEUTRONIC CHARACTERISTICS						PHYSICAL CHARACTERISTICS					
			REACTIVITY ADDITION RATES, %△K/K/min/rod					INSERTION				WITHDRAWAL	
ROD TYPE	NUMBER	WORTH EACH %△K/K	AVG.	MAX PER MIN.	MAX PER IN.	FIRST HALF	THIRD QUARTER	ACTUATOR NORMAL	ACTUATOR SCRAM	TIME NORMAL	TIME SCRAM	ACTUATOR	TIME
Safety	2	1.5	0.1	0.21	0.21	0.2	--	Motor	Pneumatic	15 min.	200 msec	Motor	15 min.
Shutdown	6	2.5	0.17	0.35	0.35	0.33	--	Motor	Spring	15 min.	0.8 sec	Motor	15 min.
tback	1	0.3	0.15	0.17	0.03	0.42	--	--	Motor	--	2 sec.	Motor	2 min
Shim	2	1.0	0.4	0.15	0.09	0.07	--	Motor	--	23 min.	--	Motor	23 min.
Fine	1	0.4	0.4	0.4	0.04	1.1	--	Motor	--	1 min.	--	Motor	1 min.
Gang I	2 Safety 2 Shutdown	8		0.54	1.12	1.12	1.06	--	--	--	--	Motor	15 min.
Gang II	4 Shutdown	10	0.67	1.40	1.40	1.32	0.9	--	--	--	--	Motor	15 min.

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356 076

CONTROL ROOM PERSPECTIVE

FIG.-14

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LETTER	DATE	CHANGE	BY	CHKD

I

FISSION CHAMBER

LINEAR PRE-AMPLIFIER

LINEAR AMPLIFIER AND DISCRIMINATOR

LOG COUNT RATE METER

LOW LEVEL PERIOD AMPLIFIER

II

FISSION CHAMBER

BINARY SCALER AND TIMER

LINEAR PRE-AMPLIFIER

LINEAR AMPLIFIER AND DISCRIMINATOR

SOURCE COUNT RATE INTERLOCK

LOG COUNT RATE METER

LOG COUNT RATE RECORDER

LOW LEVEL PERIOD AMPLIFIER

COMPENSATED IONIZATION CHAMBER POWER SUPPLY

III

COMPENSATED IONIZATION CHAMBER

LOG "N" AND PERIOD AMPLIFIER

LOG "N" RECORDER

PERIOD RECORDER

COMPENSATED IONIZATION CHAMBER POWER SUPPLY

IV

COMPENSATED IONIZATION CHAMBER

LOG "N" AND PERIOD AMPLIFIER

V

PCP IONIZATION CHAMBER

VI

PCP IONIZATION CHAMBER

SAFETY AMPLIFIER

POWER LEVEL RECORDER

VII

PCP IONIZATION CHAMBER

VIII

PCP IONIZATION CHAMBER

SAFETY AMPLIFIER

COMPENSATED IONIZATION CHAMBER POWER SUPPLY

IX

COMPENSATED IONIZATION CHAMBER

ELECTROMETER

LINEAR POWER LEVEL RECORDER

SERVO AMPLIFIER

POWER LEVEL SET POINT

MAGNETIC AMPLIFIER

REGULATING ROD MOTOR

SETBACK CKT

WATER IN MAIN LOOP

INLET DUCT TEMP HIGH

SCRAM CKT

HEAT EX. GAS OUTLET TEMP HIGH

STEAM BOILER TROUBLE

STEAM BOILER DAY TANK LEVEL HIGH

COMP OIL OR WATER RESERVOIR LOW

EXHAUST DUCT TEMP HIGH

REACTOR OUTLET GAS TEMP HIGH

MAIN LOOP FLOW 90% LOW

HIGH POWER LEVEL 110%

MANUAL

PERIOD SHORT CHAN I OR II

INST AIR PRESS LOW

T/C MANIFOLD PRESS HIGH

PERIOD SHORT CHAN III & IV

RUN-SAFE SWITCHES ON SAFE

REACTOR ROOM DOORS OPEN

LOG N HIGH LEVEL

EARTH-QUAKE

HIGH POWER LEVEL 115%

REACTOR OUTLET GAS TEMP HIGH

MAIN LOOP FLOW 85% LOW

MANUAL

POWER FAILURE

PERIOD SHORT CHAN I, II, III OR IV

SAFE RODS PRESS LOW

D-202 WASTE WATER LEVEL HIGH

BLOWDOWN N_2 SUPPLY PRESSURE LOW

MAIN LOOP OXYGEN CONTENT HIGH

COMP SEAL WATER PRESSURE LOW

ANNUNCIATOR CKT

BURNER #1 FLAME FAILURE

BURNER #2 FLAME FAILURE

BURNER #3 FLAME FAILURE

REACTOR OUTLET GAS TEMP HIGH

SAFETY ROD 1 & 2 FREE FALL LOW

TACH. SPEED LOW

PRIMARY COOLING WATER TEMP HIGH

AREA RADIATION MONITORING SYSTEM HIGH

COMPRESSOR BEARING OIL TEMP HIGH

N_2 FARM PRESSURE LOW

S-201 DEMIN. CONDUCT. HIGH

HEAT EX. SECONDARY WATER OUTLET TEMP HIGH

S-301 DEMIN. CONDUCT HIGH

WASTE N_2 STORAGE TANK LEVEL HIGH

OXYGEN STORAGE PRESSURE LOW

LOOP HUMIDITY HIGH

UTILITY AIR-LINE PRESSURE LOW

MAIN LOOP FLOW LOW

SHUTDOWN COOLING LOOP TEMP HIGH

PRIMARY COOLING WATER FLOW LOW

SHUTDOWN COOLING LOOP FLOW LOW

FISSION PRODUCT MONITOR HIGH

WATER IN RODS

REACTOR POOL LEVEL LOW

COOLING TOWER POOL LEVEL LOW

UTILITY WATER LEVEL LOW

MAIN LOOP PRESSURE LOW

SECONDARY WATER PRESSURE LOW

MAIN LOOP PRESSURE HIGH

EMERGENCY POWER SUPPLY OFF

NUCLEAR INSTRUMENT FAILURE

UTILITY WATER PRESSURE LOW

FUEL OIL PRESSURE LOW

COMPRESSOR LUBE OIL PRESSURE LOW

FUEL OIL LEVEL LOW

UTILITY WATER LEVEL HIGH

COOLING TOWER POOL LEVEL HIGH

REACTOR POOL LEVEL HIGH

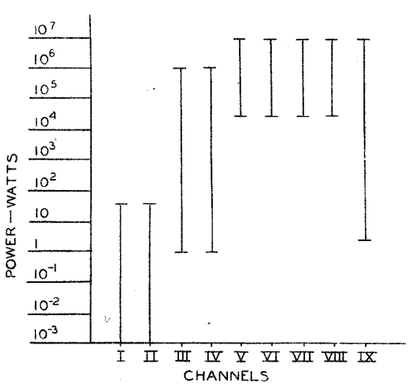
DEMIN. WATER SURGE DRUM LEVEL LOW

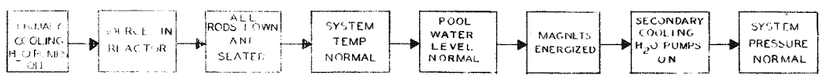
MAIN LOOP MAKE UP FLOW HIGH

DEMIN. WATER SURGE DRUM PRESSURE LOW

D-203 WASTE WATER LEVEL HIGH

FUEL ELEMENT TEMP. HIGH





REACTOR START UP INTERLOCKS

1-404000

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356 077

FIG. 15

No. Reqd.	NEXT ASSY.	MODEL	
HEAT TREAT	FINISH		
TOLERANCE UNLESS OTHERWISE NOTED			
LINEAR TOL.	.XX ± .03		
	.XXX ± .010		
ANGULAR TOL.			

INDICATES SURFACE ROUGHNESS

MATERIAL

NUCLEAR BLOCK DIAGRAM

SCALE NONE

DATE 12-10-57

DRAFTSMAN R.C.H.

RELEASE DATE

APP.

DWG. SIZE F

1-404000 PART NUMBER

AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA

SOURCE CONFIDENTIAL

DECLASSIFIED

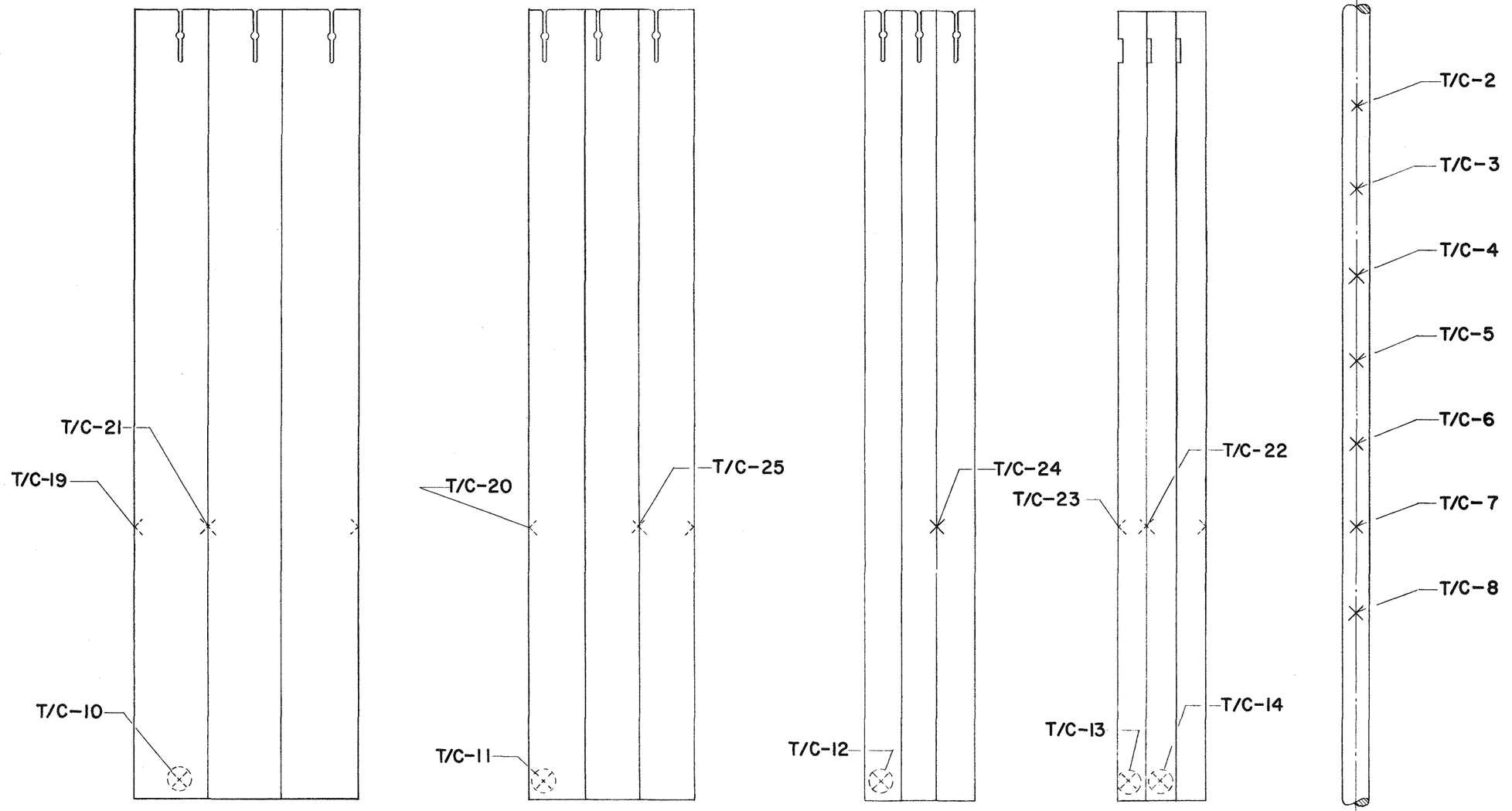
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LETTER	DATE	CHANGE	BY	CH'D	

CONFIDENTIAL

DEVELOPED VIEW OF FUEL ELEMENT PLATES SHOWING THERMOCOUPLE LOCATIONS

CONFIDENTIAL



COPY

CONFIDENTIAL

LEGEND

- INDICATES THERMOCOUPLE ON THE INSIDE OF THE PLATE OR SPINE.
- INDICATES THERMOCOUPLE ON THE OUTSIDE OF THE PLATE.
- INDICATES THERMOCOUPLE IN THE GAP BETWEEN PLATES BEHIND THE PLATE ON WHICH INDICATED.

CONFIDENTIAL

356 078

IN ADDITION TO THERMOCOUPLES SHOWN, TWO THERMOCOUPLES ARE INSTALLED IN THE INSULATION NEAR THE CENTER (VERTICALLY) OF THE ELEMENT.

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

NO. REQD.	NEXT ASSY.	MODEL	DRAFTSMAN	DATE
HEAT TREAT		FINISH	CHECKED	
TOLERANCE UNLESS OTHERWISE NOTED			STRESS	
LINEAR TOL.	.XX	± .03	ENGINEER	
	.XXX	± .010	APPROVED	
	ANGULAR TOL.		PRODUCTION	
INDICATES SURFACE ROUGHNESS FINISH PER MIL STD-10 UNLESS OTHERWISE NOTED			MATERIAL	

MATERIAL

THERMOCOUPLE LOCATIONS INSTRUMENTED FUEL ELEMENT

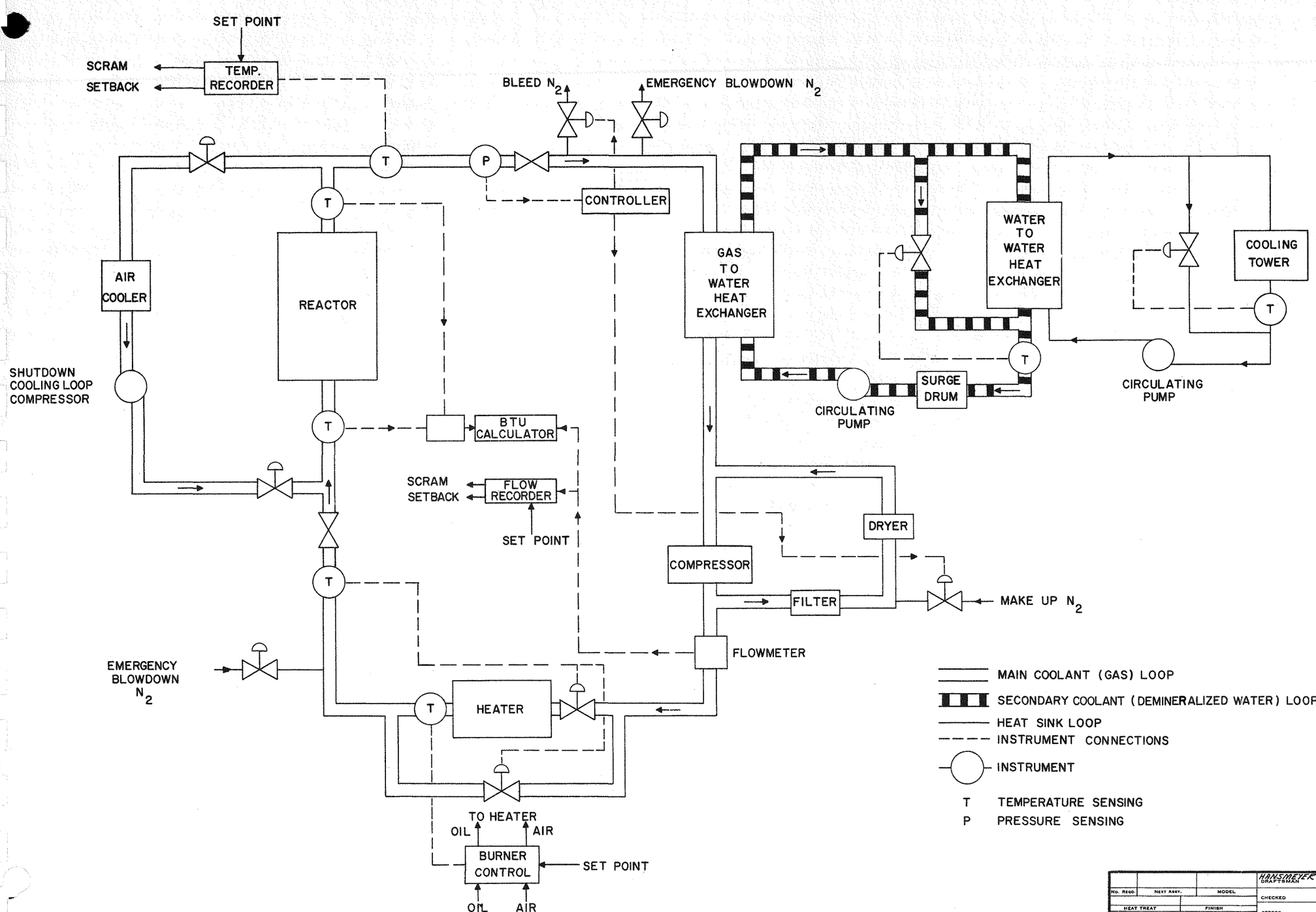
AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA

SIMILAR TO	SCALE	DWG. SIZE	FIG-16
RELEASE DATE	CAL. WEIGHT	D	DRAWING NUMBER
	ACT. WEIGHT		

DECLASSIFIED

03712AJ03 DECLASSIFIED

LETTER	DATE	CHANGE	BY	CH'D



MAIN COOLANT (GAS) LOOP

SECONDARY COOLANT (DEMINERALIZED WATER) LOOP

HEAT SINK LOOP

INSTRUMENT CONNECTIONS

INSTRUMENT

T TEMPERATURE SENSING

P PRESSURE SENSING

PROCESS FLOW DIAGRAM

FIG.-17

COPY

NOTES: 1. REMOVE ALL BURRS AND SHARP EDGES

356 079

RELEASED

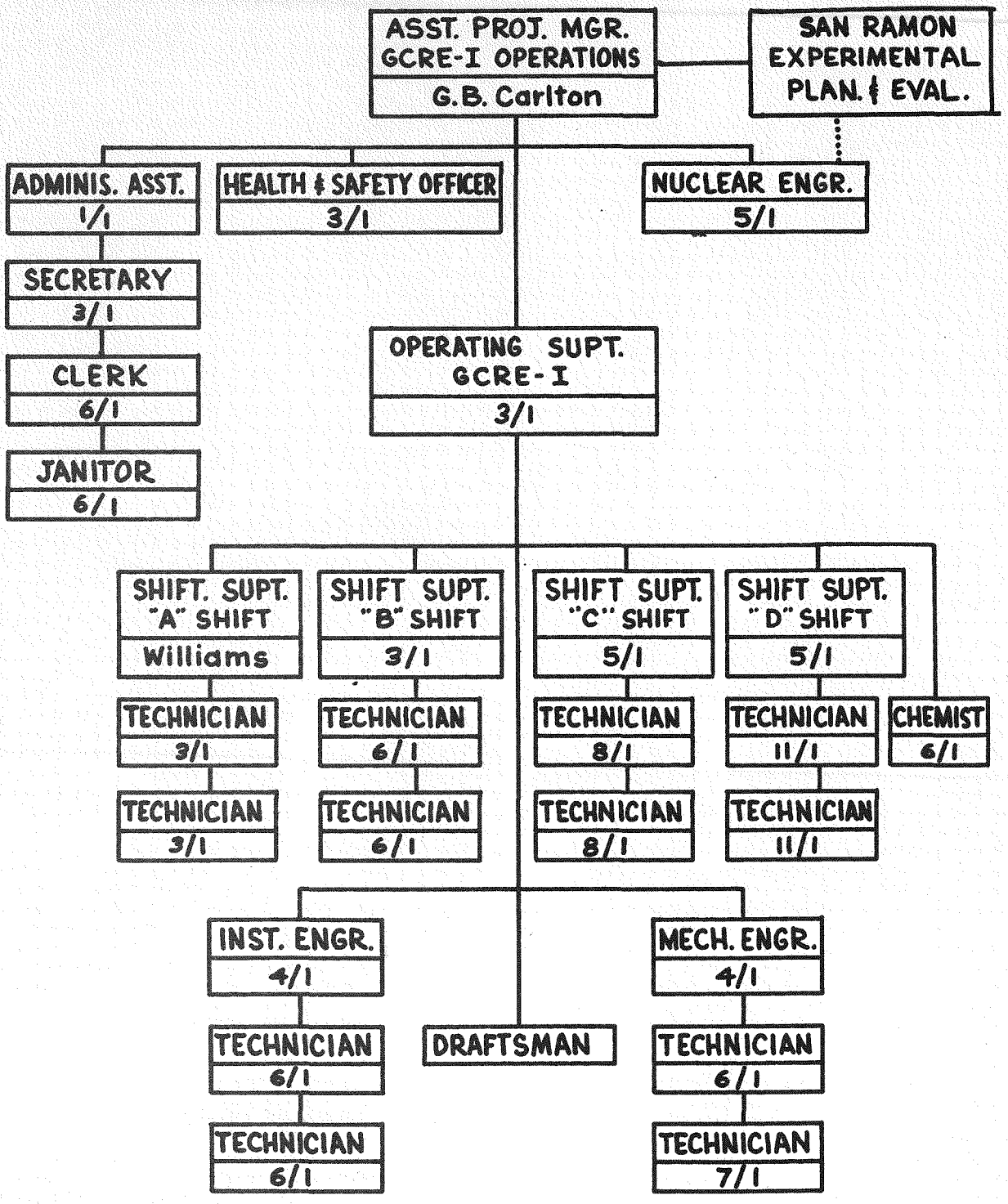
HANSMEYER DRAFTSMAN	DATE 12-22-58	MATERIAL	AEROJET-GENERAL NUCLEONICS SAN RAMON CALIFORNIA
CHECKED		UNCLASSIFIED	
STRESS			
ENGINEER			
APPROVED			
PRODUCTION		SIMILAR TO	SCALE
MATERIAL		RELEASE DATE	DWG. SIZE D
		SCAB WEIGHT	
		ACB WEIGHT	DRAWING NUMBER

No. Regd. | Next Assy. | Model | Heat Treat | Finish

TOLERANCE UNLESS OTHERWISE NOTED: LINEAR .XX ± .03; TOL .XXX ± .010; ANGULAR TOL.

INDICATES SURFACE ROUGHNESS FINISH FOR R.M.S. 63-10 UNLESS OTHERWISE NOTED

OPERATIONS ORGANIZATION



NOTES:

① 11 PROFESSIONAL or TECHNICAL PERSONNEL
② 17 HOURLY PERSONNEL
③ GUARD SERVICE PURCHASED FROM AEC-IDO

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FIG.-18

359 081

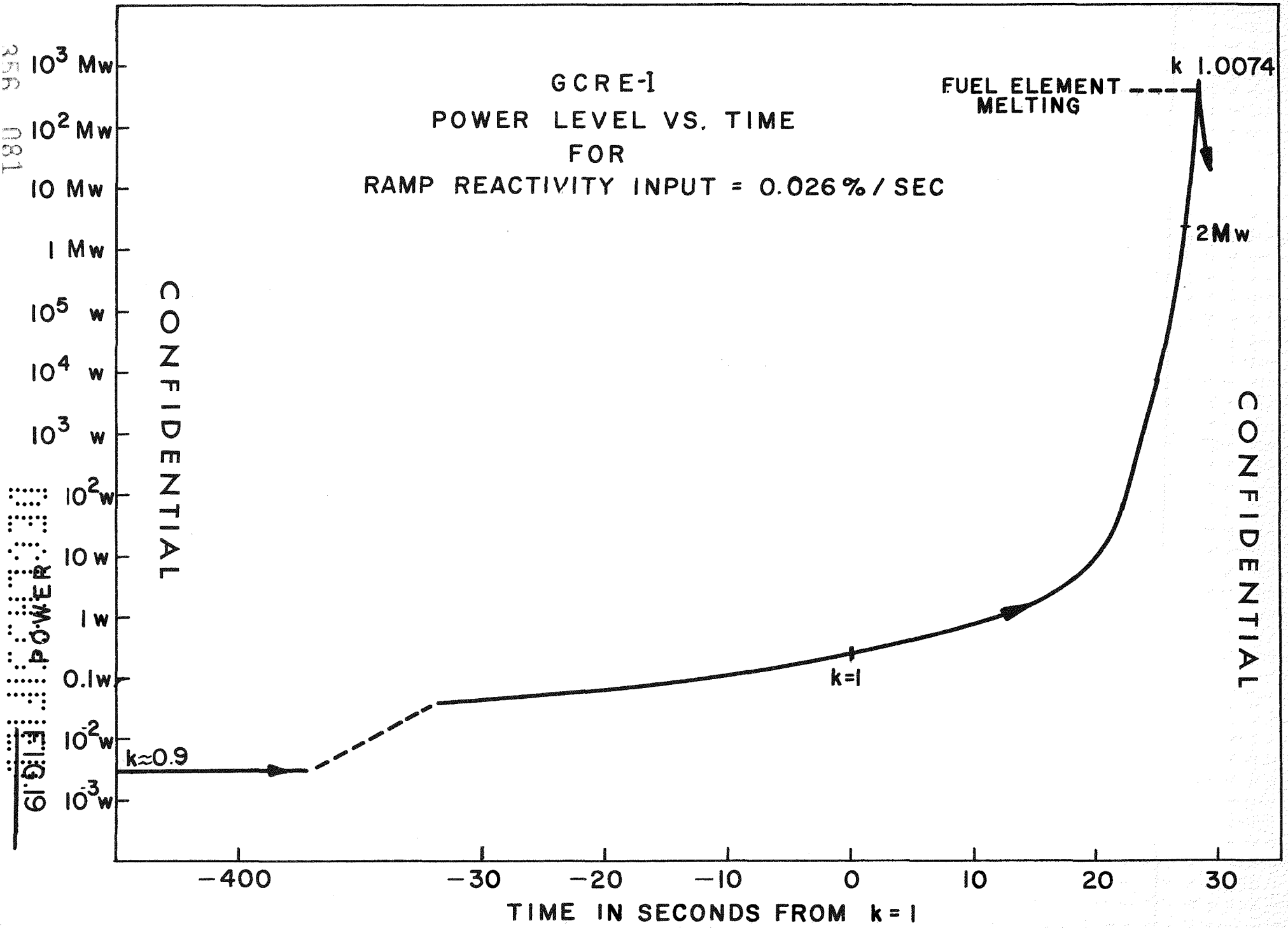


Fig. 19

356 082

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GCRE-1 RAMP RATE V.S. EQUIVALENT STEP

6%
5%
4%
3%
2%
1%

EQUIVALENT STEP

RAMP INCREASE FROM SUBCRITICAL CONDITION

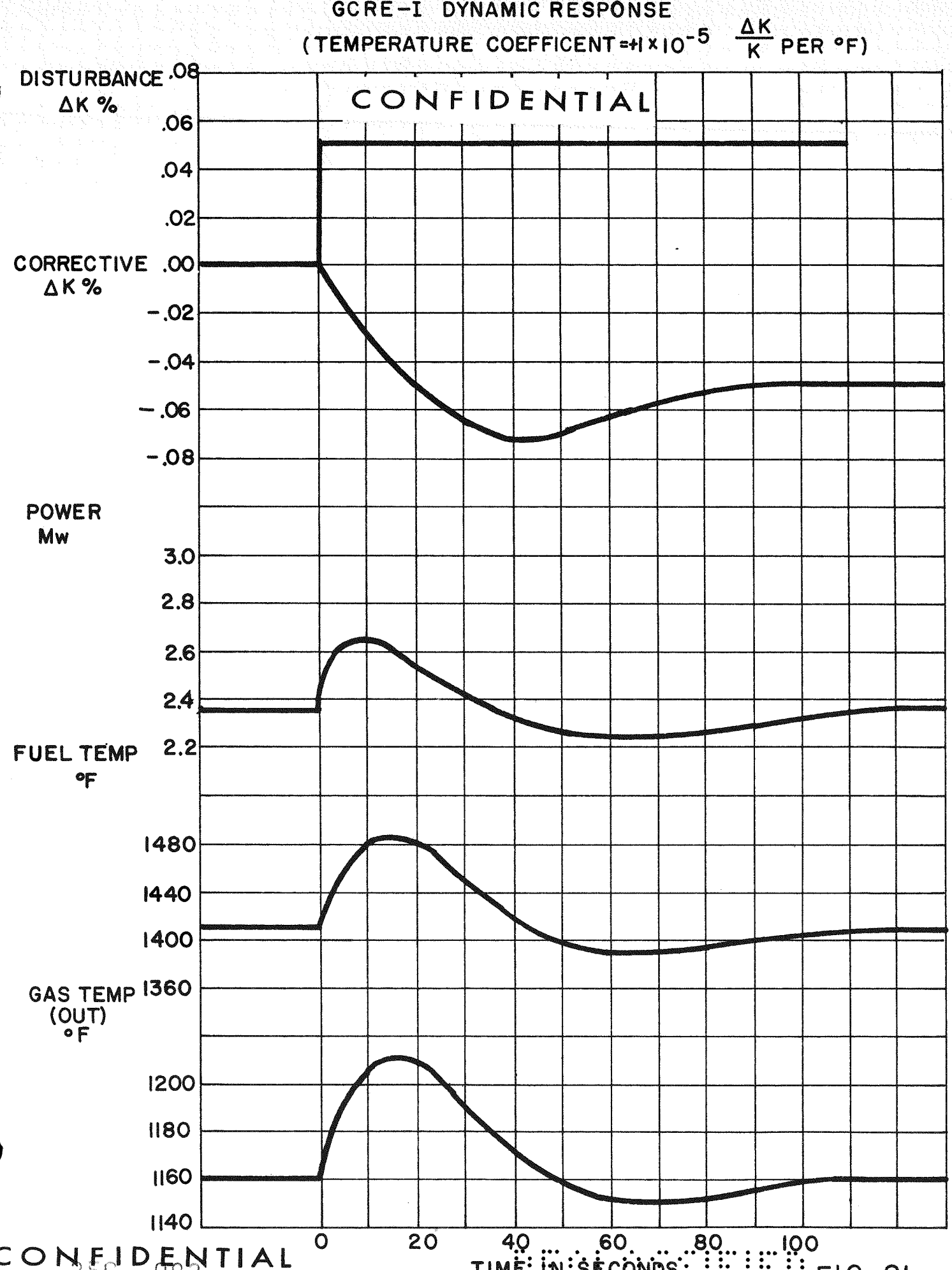
RAMP INCREASE FROM OPERATING CONDITION

.01 %/SEC.
0.1 %/SEC.
1 %/SEC.
10 %/SEC.

RAMP INCREASE IN REACTIVITY

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FIG-20



356 083

FIG.-21

356 084

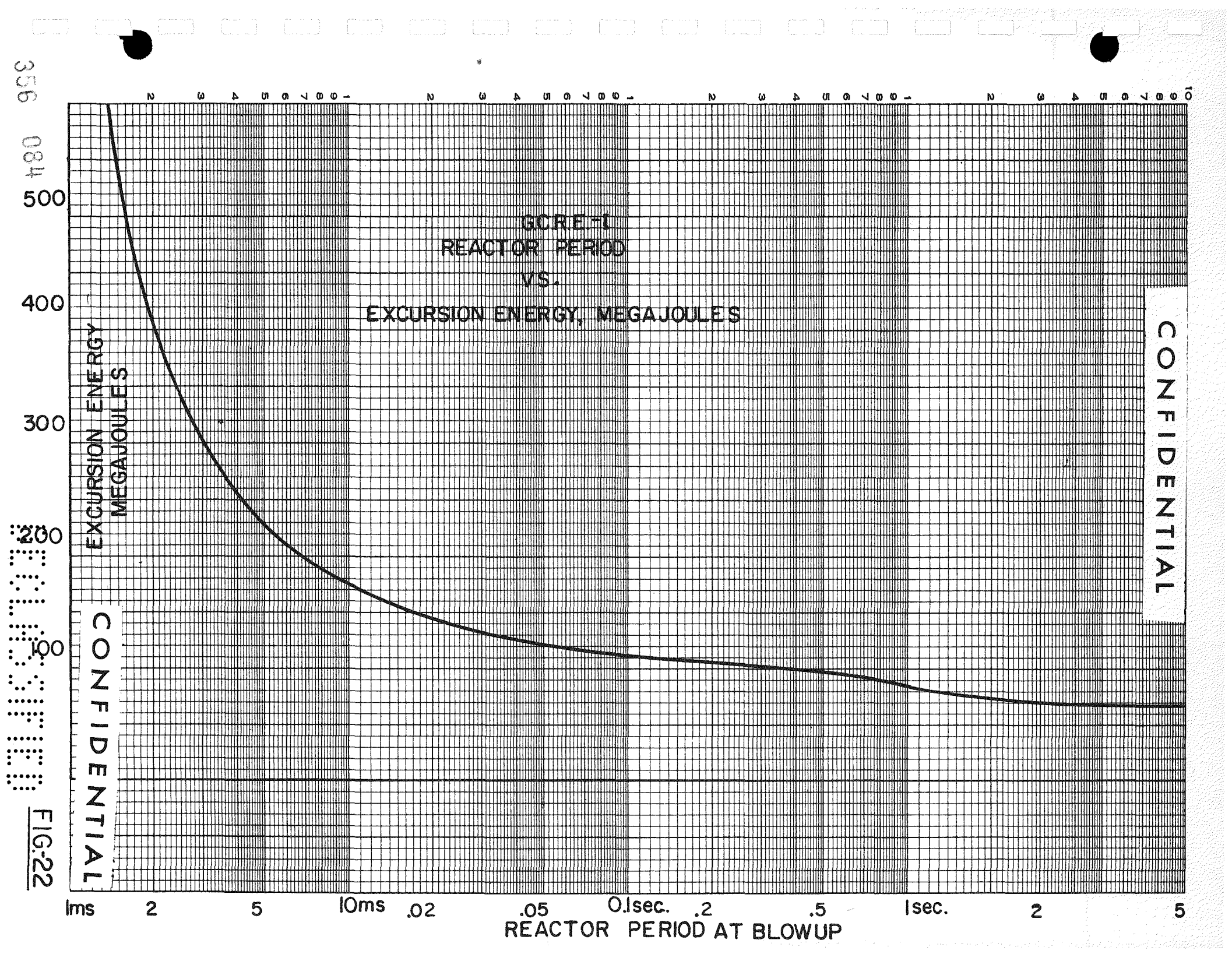


FIG. 22

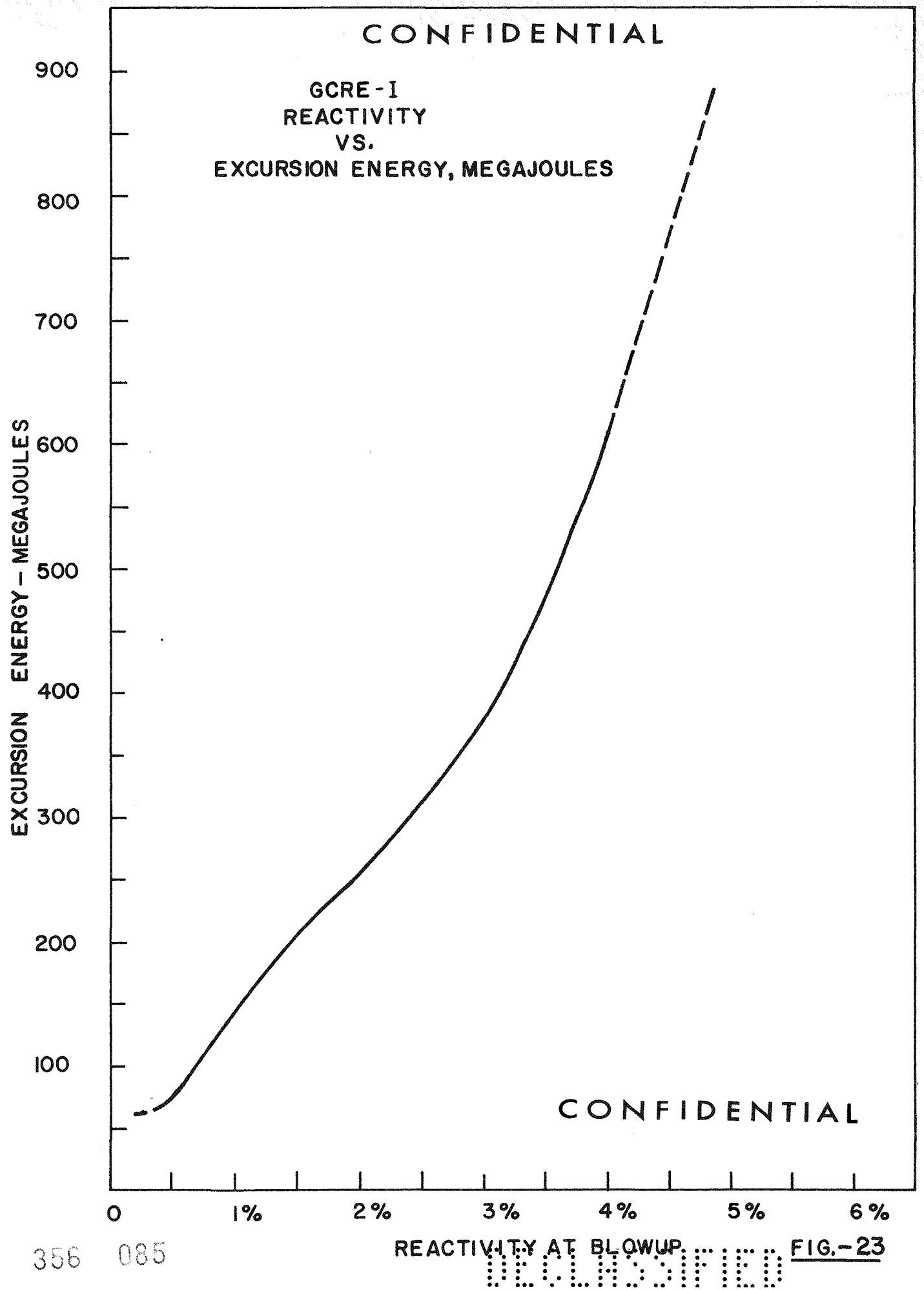


FIG.-23

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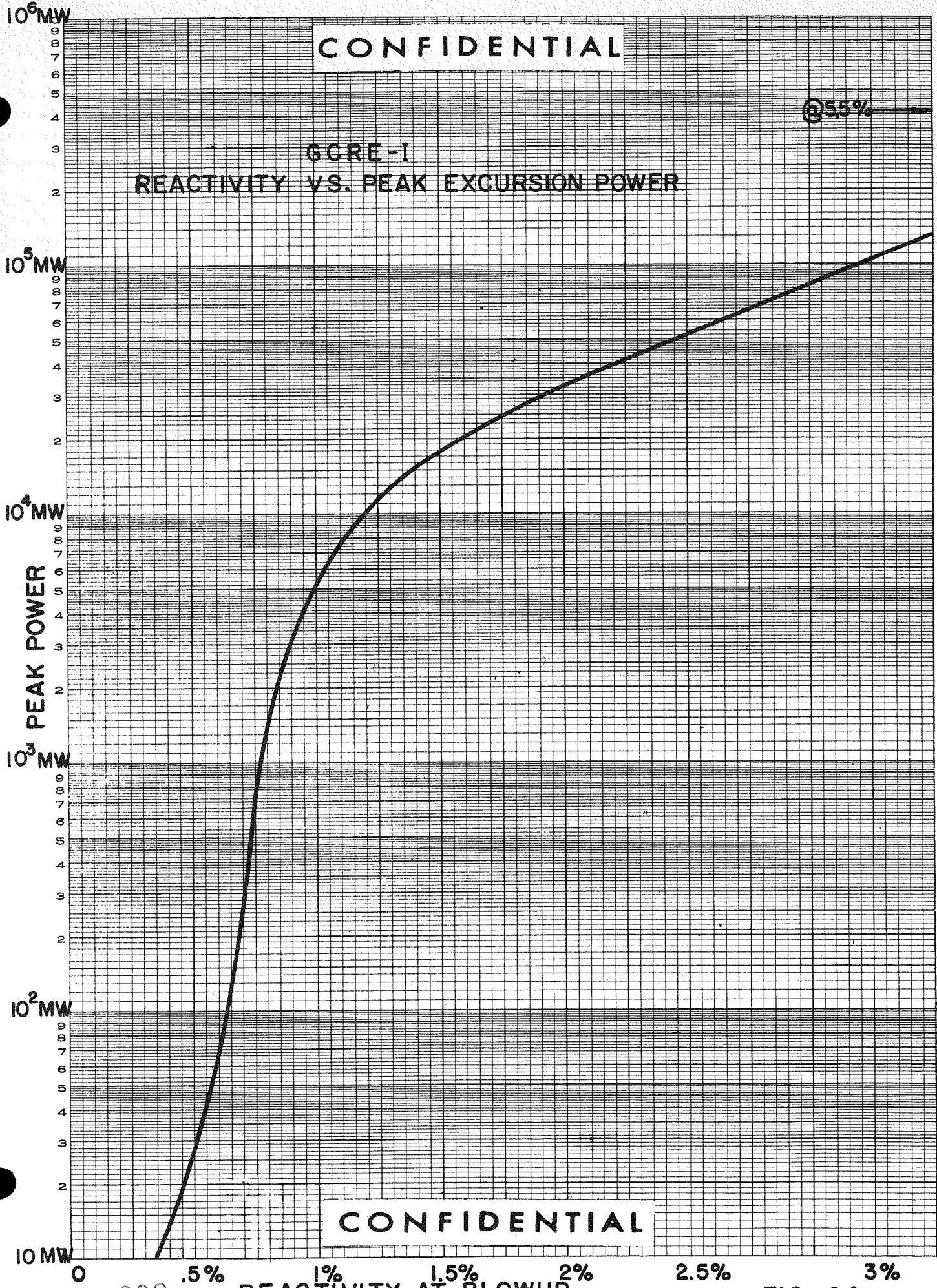


FIG-24

356 086

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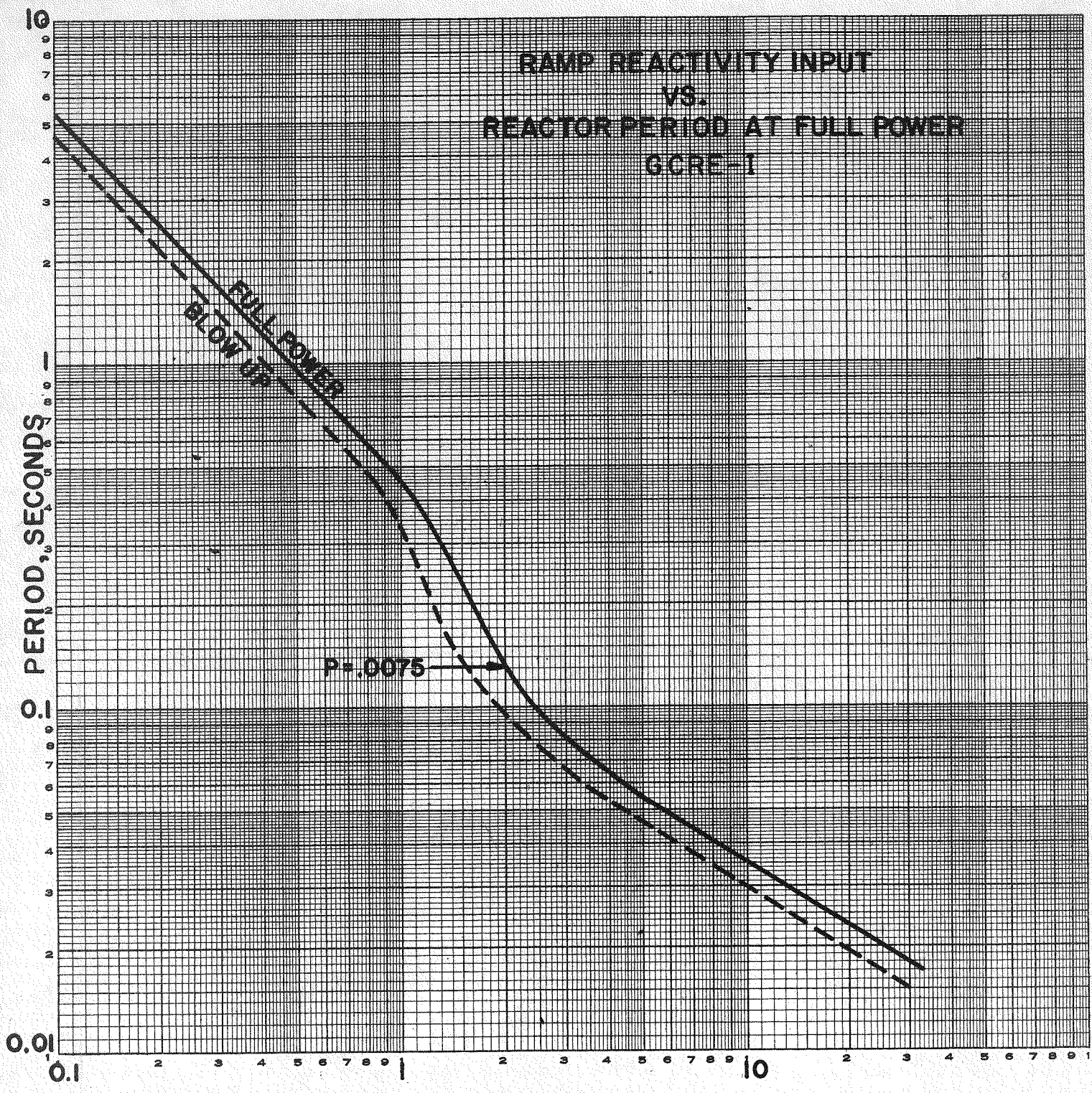


FIG.-25

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Report No. IDO-28506

APPENDIX A

REACTOR EXCURSION CHARACTERISTICS

A. INTRODUCTION

The purpose of these calculations is to examine the consequences of the introduction of various step increases of reactivity into the GCRE I reactor. The energy released during an excursion is given by

$$E = P_o \int \frac{\Delta K}{\Delta K - \beta} \exp \left\{ \frac{t}{l^*} \left[\Delta K - \beta - \Sigma \rho_n(T,t) + D(t) \right] \right\} dt$$

where $\rho_n(T,t)$ is the reactivity contribution of the various shutdown mechanisms,

$\rho_1(T,t)$ is the reactivity effect of the fuel temperature coefficient.

$\rho_2(T,t)$ is the contribution of the moderator temperature coefficient.

$\rho_3(T,t)$ is the effect of fuel element melting.

$\rho_4(T,t)$ is the contribution fuel vaporization on reactivity.

$\rho_5(T,t)$ is the result of movement of the core structure.

D is the contribution of the delayed neutron emitters.

T is temperature of the fuel elements.

t is time.

l^* is the neutron lifetime.

β is the delayed neutron fraction.

ΔK is the assumed step increase in reactivity.

Each of these mechanisms is investigated to determine the condition under which it will occur, its effect on reactivity, and the rate at which it will operate. Finite difference calculations were made of the effect of the introduction of a fixed amount of reactivity into the reactor, and the power is assumed to increase at the appropriate period until a shutdown is effected by one or more of the shutdown mechanisms.

For these calculations, the reactor core is assumed to be composed of 100-kg of stainless steel integrally mixed with 15 kg of uranium. This material is thermally insulated from the water moderator and is contained within 60 aluminum pressure tubes having 2-cm inside radius. The core is assumed to be 55-cm in diameter and 70-cm long.

The flux distribution and hot spot factors are taken from recent calculations. The neutron lifetime is 50 microseconds. Because the maximum excursion energy is obtained from an initially cold reactor, this condition is assumed. The effects of ramp inputs as compared to step inputs have been made and are shown in Figure 20.

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It is difficult to obtain a high degree of accuracy in calculations of this type, where there are a number of different mechanisms operating. These calculations do give a good indication of the magnitude of the excursion and the resulting damage but because the radiation hazard is essentially determined by the quantity of steady state fission products present in the core, the radiation hazard resulting from an excursion is nearly independent of these calculations.

B. FUEL TEMPERATURE COEFFICIENT

The temperature coefficient of the fuel in the GCRE I, as calculated by approximation, is $-1.5(\pm 1) \times 10^{-6}$ $\Delta K/K$ per $^{\circ}C$ and thus $\rho_1(T) \approx -1.5 \times 10^{-6} T$. This result is subject to errors nearly as large as its magnitude due to the fact that it is the addition of positive and negative effects (longitudinal expansion, radial expansion, and reduction of the fuel density and the disadvantage factor.) For a fuel element temperature rise of 1000°C, the reactivity change is about -0.15%, and would have only a very small effect in limiting the excursion.

C. MODERATOR PROMPT TEMPERATURE COEFFICIENT

The reactivity effect of heating the moderator has been measured to be a maximum of $+1 \times 10^{-4}$ $\Delta K/K$ per $^{\circ}C$. This coefficient is negative above about 70°C.

The moderator temperature rise during an excursion results from two effects; the fast neutron and gamma heating, and heat transfer from the fuel.

The effect of heat transfer from the fuel is $Te^{-\alpha t}$ where $\alpha = \frac{K}{\rho C_p l^2}$; the effect of the insulation between the elements and the aluminum wall alone is

$$\alpha = \frac{K}{Cpl^2} = \frac{(0.5 \text{ BTU/ft}^{\circ}\text{F hr}) \ (144 \text{ in}^2/\text{ft}^2)}{(18 \text{ lb/ft}^3) \ (0.25 \text{ BTU/}^{\circ}\text{F lb}) \ (3600 \text{ sec/hr}) \ (0.1 \text{ in})^2} = 0.45 \text{ sec}^{-1}$$

Thus it can be seen that in an excursion the heat transfer from the fuel will not contribute to the moderator temperature.

The effect of the fast neutron and prompt gamma heating on the moderator temperature, assuming 4% of the total energy generation is in the moderator, is given by

$$.04 \, M_{fuel} \, Cp_{fuel} \, \Delta T_{fuel} = M_{mod} \, Cp_{mod} \, \Delta T_{mod}$$

or

$$\Delta T_{mod} = \frac{.04 \left(\frac{M_f}{M_m} \right) \left(\frac{Cp_f}{Cp_m} \right) \Delta T_f}{}$$

$$\Delta T_{mod} \approx (.04) \ (1.4) \left(\frac{.15}{1} \right) \Delta T_f$$

$$\rho_2 = +1 \times 10^{-4} \Delta T_m = +1 \times 10^{-4} (.008) \Delta T_f$$

$$\rho_2 = 0.8 \times 10^{-6} \Delta T_f, \text{ maximum}$$

In a 400 megajoule excursion, the moderator is heated about 50°C and the total reactivity change is about + 0.15%. This effect is also small, compared to the other reactivity changes in an excursion, and will be neglected.

D. EFFECT OF FUEL ELEMENT MELTING

The physical removal of the fuel element material from the core, either as a result of fuel element meltdown or vaporization has a large effect on reactivity. It is assumed, in the meltdown or vaporization that the uranium and the stainless steel will remain integrally mixed. The removal of the fuel element material will result in a decrease in the thermal utilization. The total neutron absorption, Σa, may be taken as being composed of three parts:

Σu = uranium absorption
Σss = absorption of stainless steel in fuel elements
$\Sigma m = \Sigma a - \Sigma u - \Sigma ss$ = absorption in material other than fuel elements.

The change in reactivity due to the removal of the fuel element material is:

$$\Delta\rho = \frac{\Sigma ss + \Sigma m}{\Sigma a} \frac{\Delta\Sigma u}{\Sigma u} - \frac{\Sigma ss}{\Sigma a} \frac{\Delta\Sigma ss}{\Sigma ss} - \frac{\Sigma m}{\Sigma a} \frac{\Delta\Sigma m}{\Sigma m}$$

with integral mixing; $\frac{\Delta\Sigma u}{\Sigma u} = \frac{\Delta\Sigma s}{\Sigma s} = F$, the fraction of fuel removed, and

$\Delta\Sigma m = 0$ so that $\Delta\rho_3 = \frac{\Sigma m}{\Sigma a} F \approx 0.08F$.

This equation must be modified by the worth of the positions in the reactor occupied by the fuel element material so $\rho_3 = 0.08$ FW where $W = \frac{\phi\,\phi^+}{(\phi\,\phi^+)_{ave}}$.

There may be an additional effect resulting from the increased concentration of fuel in portions of the reactor, thus increasing the disadvantage factor, but for the purpose of these calculations this effect is neglected.

The above equation can then be used by assuming the fuel falls out of the reactor under the acceleration of gravity with its relative worth changing with position. For the purposes of the calculations performed here it was assumed that the viscous friction between the molten material and the still solid portions of the assembly decreased the net acceleration of the fluid to g/2 or 500 cm/sec^2. The distance moved by the fluid mass at any instant and the resultant reactivity change can then be calculated.

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In all cases, the fuel element material which will first begin to move, is that which is located at the position of maximum power generation (outside fuel-plate of the fuel element in the center region of the core.) The worth of 10% of the fuel located in this region (where the neutron flux and the statistical worth is greatest is:

$$W_c = \frac{\phi c \, \phi c^+}{(\phi \, \phi^+)_{ave}} \approx 3.4$$

where the subscript "c" refers to the center regions, and "ave" to the average over the core. Thus, for the removal of fuel from this region,

$$\Delta\rho = (3.4)\,(0.08F) = 0.27F$$

For the fuel removal from other regions, an appropriate value of $\frac{\phi \, \phi^+}{(\phi \, \phi^+)_{ave}}$ is used and the resultant $\Delta\rho$ is correspondingly less.

The energy required to melt the core is:

100 kg Stainless Steel	20°C to 1400°C :	$207 \frac{cal}{g}$ =	207×10^5 cal
15 kg Uranium	20°C to 1400°C :	$79 \frac{cal}{g}$ =	12×10^5 cal
100 kg Stainless Steel	Heat of fusion:	$140 \frac{cal}{g}$ =	140×10^5 cal
			359×10^5 cal

This corresponds to $\frac{359 \times 10^5 \text{ cal}}{.239 \times 10^6 \frac{\text{cal}}{\text{Mj}}}$ = 150 megajoules.

This represents the energy which is required to melt all the fuel elements. In an uncontrolled excursion, the fuel element in the regions of high neutron flux will melt sooner, and at a lower value of excursion energy than the average. The average flux in the center of the reactor containing 10% of the total fuel is about 2.4 times as great as the average flux in the reactor. Consequently, in an excursion, this region will be melted with an energy of about $\frac{149 \text{ Mj}}{2.4}$ = 62 megajoules. Before one-half of the total fuel is melted, the center region will be partially vaporized.

Therefore, at an excursion energy of 62 megajoules, 10% of the fuel will be molten and will be falling out of the reactor under the acceleration of gravity. This movement will result either in the actual removal of the fuel from the core, or in its accumulation in the bottom portion of the tube. The latter result will have a high self-shielding effect and thus a reduced worth. It was assumed that, if the material in the center region, for example, moved 50 cm, it would be effectively removed, with a resultant total reactivity change of 2.7% or an average rate of 0.054%.* For material removal from other regions, the $\Delta\rho$ is somewhat less, as a result of lower flux and statistical importance. The relative worth in the region i is: $W_i = \frac{\phi \, \phi}{(\phi \, \phi^+)_{ave}}$. The reactivity changes from these separate effects are approximately additive.

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* K/K per cm.

For reactor periods that are longer than about 1 sec., the removal of molten material is sufficient to shutdown the reactor before any substantial quantity of the fuel elements has vaporized. For reactor periods that are less than about 30 msec, this effect is negligible because vaporization will occur before the material has moved an appreciable distance.

E. VAPORIZATION

1. Introduction

The vaporization and subsequent ejection of the fuel will result in reactor shutdown. There are several possible mechanisms which must be investigated.

a. The escape of the vaporized material.
b. The vapor pressures in the center section of the elements forces the solid and liquid portions of the element out either end.
c. The vapor pressure bursts the pressure tubes and causes radial displacement of the moderator.

For pressures less than 3000 psi it is found that the vapor pressure forcing solid and liquid out the ends is the dominant effect. For pressures greater than 3000 psi, the radial displacement of the moderator is larger.

The following nomenclature will be used:

P	=	Pressure of vaporized material
N	=	Atom density
k	=	Boltzmann constant
T	=	Temperature
R	=	Gas constant
c	=	Sonic velocity
γ	=	Specific heat ratio
ρ	=	Density
v	=	Velocity
t	=	Time
l	=	Length
E	=	Energy
V	=	Volume

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r = Radius, from tube centerline

F = Fraction of fuel removed

F^1 = Fraction of fuel vaporized

a = Acceleration.

2. Gas Description

The energy required to vaporize the core is:

Total to melt stainless steel			359×10^5 cal
100 kg stainless steel	1400°C to 2500°C	200 $\frac{cal}{g}$	200×10^5 cal
15 kg uranium	1400°C to 2500°C	42 $\frac{cal}{g}$	6×10^5 cal
Vaporize stainless steel		1500 $\frac{cal}{g}$	1500×10^5 cal
	Total		2065×10^5 cal

and

$$\frac{206.5 \times 10^6 \text{ cal}}{.239 \times 10^6 \frac{\text{cal}}{\text{MJ}}} = 864 \text{ megajoules.}$$

For the fuel in the center region, with relative worth of 2.4, vaporization will start at 98 megajoules and will be complete at 360 megajoules.

If the fuel elements are completely vaporized, the resultant pressure is roughly $P = NkT^*$.

The average density in the gas channel is about $2 g/cm^3$, and the atomic number is about 56, so that the atom density is:

$$N = \frac{6 \times 10^{23}}{56} \times 2 = 2.15 \times 10^{22}/cm^3$$

and

$$P_o = (2.15 \times 10^{22}/cm^3)\ (1.38 \times 10^{-16} \frac{ergs}{°K})\ (2770°K)$$

$$= 8.2 \times 10^9 \text{ dynes/cm}^2$$

$$= 120{,}000 \text{ psi}$$

* Near the vaporization temperature, NkT is a substantial overestimate of pressure. However, in an excursion, the pressure of the vapor will increase rapidly and approach NkT as a limit.

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The sonic velocity at vaporization temperature will be:

$$c_o = \sqrt{\frac{\gamma P}{\rho}} = \sqrt{\gamma RT}$$

and for $\gamma = 5/3$, $c_o = 8.3 \times 10^4$ cm/sec.

3. Axial Movement

A rarefraction will proceed, axially, into the vaporized region with sonic velocity and the material at the ends will be blown out with a wave front velocity of $v = \frac{2}{\gamma - 1} c \approx 3c_o$.

However, not all of the material at a distance (ct) into the tube, will have blown out at time (t). Distance (ct) is the distance beyond which nothing has blown out or, in fact, has moved at all. In an adiabatic expansion, the velocity of matter out each end is $v = \frac{2}{\gamma + 1} c_o$

and its density is $\rho = \rho_o \left[1 - \frac{\gamma-1}{\gamma+1} \right]^{\frac{2}{\gamma-1}}$

where ρ_o and c_o are initial conditions, ($\rho_o = 2$ g/cm^3) and the flux out each end for $\gamma = 5/3$ is $\rho v = \rho_o c_o \left(\frac{2}{\gamma-1} \right)^{\frac{\gamma+1}{\gamma-1}} \approx 0.3 \rho_o c_o$.

The area density is $\rho_o l$ where l is the length of the region so that the fraction F which is removed in time (t) is $F = \frac{2 \times 0.3 \rho_o c_o t}{\rho_o l} = \frac{0.6 c_o t}{l}$

and to remove a fraction, say, F = 0.2 of a 70 cm element which is completely vaporized, would require $t = \frac{0.2 \times 70}{0.6 \times 8.3 \times 10^4} \approx 0.3$ milliseconds.

It is shown later, however, that for a complete vaporization, most of the gas energy is transmitted radially into the moderator in a shock wave. For a partial vaporization where the tube walls remain intact (P < 3000 psi), material will be ejected from the ends by two means:

(i) The escape of vaporized material
(ii) The vapor pressure in the center will force out the solid and liquid portions of element on either end.

These are examined and compared:

(a) Vaporized Material

At vaporization temperature, $c_o \approx 8.3 \times 10^4$ cm/sec and ρ_{gas} is nearly proportional to fractional quantity (F') of the fuel element

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which has been vaporized, so that before expansion, gas = $F'\rho_o = 2F'$ g/cm^3 and where the tube walls are intact, the fraction of the fuel removed is:

$$F = \frac{0.6\,\rho_{gas}\,c\Delta T}{\rho_o l} = 0.6\,\frac{(2F')\,c\Delta t}{\rho_o l} = 710\,F'\Delta t$$

where, $P < 3000$ psi $= 2 \times 10^8$ dynes/cm^2

or, $F' < 0.025$ and at the limiting value, tube rupture, $\frac{dF}{dt} \approx 0.18$/sec.

It is concluded that where the amount of vaporized material is small, the amount of escaping vaporized material is likewise small, and therefore the reactivity controlled is small.

b. Solid and Liquid Material

While the pressure of vaporization is building up in the center regions of the element, it will exert a static force on the solid and liquid portions at each end of the fuel element tube. (Most of a given fuel element will be melted before the vaporization starts at the center and thus the element will have no structural strength). The area density of one-half a fuel element is:

$$2\ \text{g/cm}^3 \times 35\ \text{cm} = 70\ \text{g/cm}^2$$

and the resultant acceleration resulting from the pressure is, $a = \frac{d^2x}{dt^2} = \frac{P}{70}$ cm/sec^2. $P \approx 8.2 \times 10^9\ F'$ dynes/cm^2 and $\frac{d^2x}{dt^2} = 1.2 \times 10^8\ F'$ cm/sec^2.

Taking the limiting case again, $F' = .025$. The time required to remove $F = 0.18$ of the solid material is about 2 milliseconds; compared to 1 second for the escape of the gas. The ejection of solid and liquid material is thus the primary shutdown mechanism at $P < 3000$ psi. At $P > 3000$ psi the energy transfer into the moderator becomes dominant, as will be shown in the next sections.

In calculating the amount of material ejected, it was assumed that as a result of the condensation and the frictional effects, the static gas pressure was but one-half effective in forcing the material out of the ends, so that initially $\frac{d^2x}{dt^2} = 0.6 \times 10^8\ F'$ cm/sec^2. If no more vapor were being generated, this acceleration would decrease as the gas expanded with $P = P_o \left(\frac{x_o}{x}\right)^{\gamma}$.

The effects of gas generation and expansion must be considered. For $P < 3000$ psi, a finite difference calculation of Δx may be made, using $\frac{d^2x}{dt^2} = 0.6 \times 10^8\ F'\ (t) \left(\frac{x_o}{x}\right)^{5/3}$ and the resultant Δx converted to reactivity by the relations derived previously.

4. Energy Distribution After Total Vaporization

The energy out the tube ends is given by:

$$E_{end} = \tfrac{1}{2} (v\ gas)^2\ \rho_{gas}\ v_{gas}\ 2\pi r^2\ \Delta t$$

where $v_{gas} = \dfrac{2}{\gamma_{gas} + 1} c_o$

and $\rho_{gas} = \rho_o \left(\dfrac{2}{\gamma_{gas} + 1} \right)^{\frac{2}{\gamma_{gas} - 1}}$

The energy from the vaporized material transmitted to the water moderator is:

$$E_{mod} = P_o\ 2\pi r l\ \Delta r$$

$$= P_o\ 2\pi r l\ v_{mod}\ \Delta t$$

where $v_{mod} = \sqrt{\dfrac{2}{\gamma_{mod}+1} \dfrac{P_o}{\rho_{mod}}}$

and initially

$$\frac{E_{end}}{E_{mod}} = \frac{r}{2\ l} \frac{\left(\dfrac{2}{\gamma_{gas} + 1} c_o \right)^3 \rho_o \left(\gamma_{gas} + 1 \right)^{\frac{2}{\gamma_{gas} - 1}}}{P_o \left(\dfrac{2}{\gamma_{mod} + 1} \dfrac{P_o}{\rho_{mod}} \right)^{\frac{1}{2}}}$$

For $\gamma_{gas} = 5/3$

$\gamma_{mod} = 3$, at this pressure

$P_o = 8.2 \times 10^9$ dynes/cm^2

$c_o = 8.3 \times 10^4$ cm/sec

$\rho_{mod} = 1$ g/cm^3

$\rho_o = 2$ g/cm^3

$$\frac{E_{end}}{E_{mod}} = \frac{1}{5 - 3} \frac{r}{1} = \frac{1}{53} \frac{2}{70} = \frac{1}{200}$$

Thus, with total vaporization, only 1/200 of the energy is directed out the tube ends.

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This relationship is also true for partial vaporization, provided that the rate of vaporization is sufficiently rapid to send a shock wave into the moderator, and indicates that the radial displacement of the moderator is the dominant effect for rates of vaporization of this magnitude.

5. Radial Displacement of Moderator

For high values of F' (the vaporized fraction) most of the energy of the vaporized fuel is transmitted to the moderator and the effect of this energy transfer on reactivity must be investigated. This is done for the case of complete fuel element vaporization.

An enormous pressure is exerted sideways through the channel walls into the moderator. This pressure causes the interface to move and sends a shock radially into the moderator. The shock velocity is:

$$v = \sqrt{\frac{\gamma + 1}{2} \frac{P}{\rho}} = \sqrt{\frac{3+1}{2} \frac{(8.2 \times 10^9)}{1}}$$

$$= 1.28 \times 10^5 \text{ cm/sec, and the material velocity}$$

$$v = \sqrt{\frac{2}{\gamma + 1} \frac{P}{\rho}} = 0.64 \times 10^5 \text{ cm/sec.}$$

This is the initial velocity. As the wall moves out, the gas expands to a larger volume. However, during an excursion, heat will continue to be generated behind the shock front. Therefore, the shock velocity which varies as $\sqrt{T}$ will be but slightly reduced with expansion. The energy is transmitted into the moderator in a time of about $\frac{2.5 \text{ cm}}{10^5 \text{ cm/sec}} =$ 0.025 msec.

Because most of the energy goes into the moderator, the time required to eject material axially is increased. After .025 msec the energies contained in the gas and the moderator are roughly proportional to their volumes. The gas volume is 0.4 of the total, and its energy density is about 0.4 of what it was originally. When the gas gives up its energy radially, some of the gas may condense. That part which does condense will not exert an axial pressure and the shutdown mechanism described in (3) will be reduced. If this phenomenon occurs, the disassembly will be governed by the time required for the entire core to blow apart, because of the increased time required for the axial removal of the fuel.

The explosive nature of the excursion is not reduced by the condensation phenomenon. The condensation has occurred only because of the transfer of the energy in the gas to the moderator in the form of a shock wave which will explosively disassemble the reactor.

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It has been shown that the moderator absorbs 0.6 of the energy in the fuel channel in less than 0.1 msec. Half of this energy will be kinetic, the other half will be dissipated as heat, behind the shock front, thus raising the moderator temperature. The initial gas energy is approximately:

$1750 \text{ cal/g} \times 2 \text{ g/cm}^3 = 3500 \frac{\text{cal}}{\text{cm}^3}$ and the moderator temperature rise is

$$T = \frac{(3500) \frac{\text{cal}}{\text{cm}^3} (\frac{1}{2}) (0.4)}{1 \text{ cal/cm}^3 \, ^\circ\text{C}} = 700^\circ\text{C}.$$

The static pressure in the core, with half of the energy going into the heat, is:

$$P = (\tfrac{1}{2}) (0.4) \left| 8.2 \times 10^9 \right| \frac{\text{dynes}}{\text{cm}^2}$$

$$= 1.64 \times 10^9 \frac{\text{dynes}}{\text{cm}^2}$$

$$= 24000 \text{ psi}$$

Behind the shock front, the core will now be blown apart with the material velocity of 6×10^4 cm/sec. This movement, starting first in the center region of the reactor, reaches the outer diameter in less than 1/2 msec. The increase in fast-neutron leakage will shutdown the reactor because:

the buckling, $B \sim 1/r$ and the age, $\tau \sim \frac{1}{\rho^2} \sim r^6$

then $B^2 \tau \sim r^4$ for uniform moderator density.

For an excess reactivity of .02 (for example) $1+B^2\tau$ must increase by .02, $B^2\tau$ must increase $.02 \frac{(1 + B^2\tau)}{(B^2\tau)} = 0.08$ so that r must increase by about $\frac{.08}{4} = .02$ or 1/2 cm in order to shutdown the reactor.

A lead reflector surrounds the outer diameter of the core, but a three inch space is left at the top and bottom for free water convection. As the shock wave reaches this high density lead, most of the energy will be reflected back into the core. A high static pressure will be maintained in the core which will move the lead, but at a rate much less than the material velocity of water. For the pressures which will be present, the time required to move the lead a sufficient distance to shut the reactor down is about 1 msec after the shock wave reaches the lead.

Simultaneously water will be forced out the 3-in. spaces above and below the lead with a velocity of 6×10^5 cm/sec. The time required to remove sufficient water to shutdown the reactor is about 0.8 msec (the time required for the shock wave to travel to the outer corners of the

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reactor) and is nearly independent of the reactivity change required. Thus the primary shutdown mechanism under these conditions is removal of water rather than movement of lead reflector.

Because the shock and material velocities vary as $\sqrt{P}$, and P varies directly with F' (the fractional vaporization), the shutdown time for the radial displacement of the moderator will vary as $1/\sqrt{F'}$. Because of the uncertainties in the shock wave calculations as a result of the non-homogeneity of the shock propagating medium, it was conservatively assumed that with material vaporization of 100% of the fuel elements in the center region, a shutdown would require 1 msec; so that, shutdown time = $1\sqrt{F'}$ msec was taken to apply from F' = .025 (tube rupture) to F' = 1. The additional shutdown effects due to axial displacement of fuel, movement of lead reflector, and movement of aluminum end reflectors was disregarded.

F. EXCURSION CALCULATIONS

Using the shutdown mechanisms which have been described it is possible to construct a detailed description of the reactor shutdown for any given reactivity. An example is given below:

For reactivity $\rho = .0076$

$\omega = 10\ \text{sec}^{-1}$

$T = \frac{1}{\omega} = 0.1$ sec.

Shutdown is assumed to start when the center section is melted, $\int P dt = 62$MJ, P = 620 Mw and subsequently:

Time Sec	ρ	$T=1/\omega$ Sec	P MW	P Δt MJ	$\int P dt$ MJ	$\Delta\rho_1$	$\Delta\rho_2$	$\Delta\rho_3$
0	.0076	0.10	620					
				6.5	62	0	0	0
.01	.0076	0.10	685					
				7.2	68.5	0	0	0
.02	.0076	0.10	757					
				7.9	75.7	.0001	0	0
.03	.0075	0.13	822					
				8.5	83.6	.0002	0	0
.04	.0074	0.15	880					
				6.0	92.1	.0002	.0001	0
.047	.0073	0.18	917	2.7	98.1	.0003	.0001	.0001 vaporization starts
.050	.0071	0.25	931	1.9	100.8	.0004	.0001	.0049
.052	.0022	-.50	934	1.7	102.7	.0004	.0001	.0109
.054	-.0038	-.006	760	1.3	104.4	.0004	.0002	.019
.056	-.012	-.003	490	0.8	105.7		.0002	.027
.058	-.020	-.004	280	0.5	106.5		.0002	.035
.06	-.028	-.002	180	1.5	107		.0003	.040
.07	-.033	-0.1	140		108.5			

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$\Delta\rho_1$ Is the result of molten fuel dropping from the center section

$\Delta\rho_2$ Is the result of molten fuel dropping from a region surrounding the center section.

$\Delta\rho_3$ Is the result of the ejection of molten and solid fuel out the ends of the center elements, by the pressure of vaporized material.

The total excursion energy for this example was found to be 110 megajoules; the peak power to be 934 megawatts. Shutdown was primarily the result of the vaporization of a small fraction of fuel element material in the center regions of the reactor. About 1/3 of the total fuel element material is melted. In about 0.5 sec additional time, this material will have fallen out of the reactor, thus bringing the reactor to a condition of very low multiplication.

Similar finite difference calculations were made for various values of reactivity ranging from ρ = .003 (T = 10 seconds) to ρ = .055 (T = 1 millisecond).

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APPENDIX B

EXPLOSION RESULTING FROM A MAXIMUM ACCIDENT

A. BASIS FOR CALCULATIONS

In this analysis a maximum accident is considered to be a reactor excursion which results in the generation of 400 megajoules of fission energy. In this excursion approximately one-fourth of the fuel-element material is vaporized and most of the remainder is melted. The enormous vapor pressure of vaporized stainless steel will cause the explosive disassembly of the reactor. In addition to the fission energy released, a chemical energy potential exists from metal-water reactions and reactions involving the evolved hydrogen.

1. Fission Energy

The amount of kinetic energy contained in the vaporized fuel element material that is available to cause an explosion is:

$$E_{ex} = 3/2\ NkT$$

where

N = number of atoms = $\frac{25 \times 10^3 \times 6 \times 10^{23}}{56}$

k = Boltzmann constant = 1.38×10^{-16} erg/^{o}K

T = Temperature = 2770^{o}K

so that

E_{ex} = 15 megajoules

The remaining 385 megajoules is dissipated as heat within the reactor core.

2. Energy from Metal-Water Reaction *

There is approximately 400 megajoules of potential energy available from the chemical reactions between the core metals and the water-moderator. An analysis of the possible reactions is given in AEC Research and Development Report IDO-28000, "The Reactions of Metals in Oxidizing Gases at High Temperatures," dated 30 April 1957, published by Aerojet-General Corporation. This report describes experiments that were made and gives the data that were found. Included in this report are the reactions that resulted from the spraying of molten metal into water and from the dropping of molten metal drops into water.

Because the core uranium is in an oxide form, the only metal-water reactions considered here are stainless steel-water and aluminum-water.

*Approximately 30 kg of zirconium has been added to the core in the form of the fuel element outer liner since issuance of the initial report. The zirconium-water reaction will contribute to the total energy release and to the maximum shock energy; however this contribution is not large enough to significantly alter the consequences of the energy release: (~ 10 MJ addition to total energy and 0.77 MJ to shock energy)

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a. Stainless Steel - Water Reactions

There is approximately 100 kg of stainless steel in the core. In an uncontrolled reactor excursion about 25% of this material will be vaporized and most of the remainder will be melted. If all the 25 kg of vaporized stainless steel reacted with water, the energy released would be:

$$E = \frac{(25 \times 10^3 \text{ g})\ (118 \text{ cal/g})}{.239 \times 10^6 \text{ cal/MJ}} = 12.3 \text{ megajoules}$$

Of this energy, less than 10% is available as kinetic energy in a form which will cause an explosion. The remaining energy is dissipated as thermal energy within that core. (In the referenced report, the highest measured conversion of liberated energy into shock energy was 7.7%, using zirconium as the test material. Using stainless steel as the test material, the conversion was 2.5%).

The maximum shock energy released by the reaction of the vaporized stainless steel with water is calculated to be one jegajoule.

The 75 kg of molten stainless steel will be in the form of relatively large drops. In general, metal-water reactions which involve particulate metal do not produce high-order explosions (taking milliseconds rather than microseconds to attain peak pressure.) These reactions can be described as deflagrations instead of detonations.

When the stainless steel particulates in a spray are of a mean diameter of about 1000 microns, less than 50% of the material will react with water.

A spray test was made using molten stainless steel and resulted in the shock energy release of but 2.5% of the potential. This reaction value is significantly reduced when the droplets are larger than 1000 microns. Therefore, while the calculated potential energy is 37 megajoules, the actual energy released will be less than 20 megajoules of which shock energy will be less than 1 megajoule.

b. Aluminum-Water Reactions

In the referenced report (IDO-28000) it is shown that, at temperatures of less than 1170°C, the reactivity of aluminum with water is nil. The aluminum is in direct contact with the water-moderator and it is thermally insulated from the fuel elements. Therefore, it is unlikely that large portions of the aluminum will be heated to temperatures greater than 1170°C in a 400 megajoule reactor excursion.

At localized portions of the core, the aluminum could be heated above 1170°C either by jets of molten stainless steel or by the penetration of stainless steel fragments. If all the aluminum in the core (25 kg) reacted with the water, the energy thus released would be

$$E = \frac{25 \times 10^3 \text{ g} \times 3550 \text{ cal/g}}{.239 \times 10^6 \text{ cal/MJ}} = 370 \text{ megajoules}$$

an energy release which is comparable to the fission energy release. In tests which were made to determine the reactivity of molten aluminum with water, it was found that the highest release of shock energy was less than 1% of the potential energy.

Therefore, if all the core aluminum, in the form of a molten spray, reacted 100% with the water-moderator the resultant release of shock energy would be less than 3.7 megajoules. The actual release of shock energy would, however, almost certainly be less than 1 megajoule.

c. Conclusions

On the basis of the above analysis, it is believed that the maximum shock energy released by a 400 megajoule reactor excursion will be 3 megajoules. The total energy release would be about 100 megajoules. This energy would be the result of the oxidation of 35 kg (1/3) of the stainless steel in the fuel elements, and 5 kg (1/5) of the aluminum in the pressure tubes. The uranium is in an oxide form, and the tube headers are of massive construction and remain virtually unheated in an excursion so that both of these materials have been disregarded in this analysis.

3. Hydrogen Reaction

The total amount of hydrogen gas which will be produced by the metal-water reactions is calculated to be 1.8 kg. The energy that is available from the reaction $H_2 + 1/2\ O_2 \longrightarrow H_2O$ would be 275 megajoules. The hydrogen gas is generated under water where there is essentially no oxygen gas present. Therefore, it is assumed that any hydrogen explosions which occur will do so after explosions resulting from fission and/or chemical energy releases. Consequently, the pressures from fission and/or chemical energy explosions and pressures from hydrogen reactions are not additive.

A hydrogen explosion will occur only when H_2 is combined with air into an explosive mixture and is detonated. The quantity of air which could be contained in the plenum chambers would not be sufficient to oxidize more than a small fraction of the available hydrogen. The most likely place for an explosive mixture of hydrogen and air to collect is in the space above the pool. At standard conditions the 1.8 kg of hydrogen will occupy a volume of 700 cu ft. The enclosed space above the pool has a volume of over 10^5 cu ft. The minimum concentration of hydrogen in air, to form an explosive mixture, is about 18%. At equilibrium, the hydrogen contained in the air above the pool will be less than 1% and well below the flammability point (approximately 4%).

Although the hydrogen-air mixture immediately above the pool and shortly after the excursion may be within the explosion range, it will be accompanied by a large amount of water vapor. This water vapor will tend to reduce both the possibility and the severity of explosion. The occurrence of an explosive mixture

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above the pool will be a transient condition, and only a small fraction of the hydrogen would be contained in an explosive mixture with air at any particular instant. Therefore, while the potential energy of the hydrogen evolved from a maximum accident is nearly equal to the initial excursion energy, the probability of its detonation in a single explosion is very remote. Any reaction which does occur would probably consist of small areas of inflamation and/or detonation at the pool surface. The localization of the reactions will not cause gross damage to the building.

4. Pressures Exerted on Pool Walls

The total shock energy released by a 400 megajoule reactor excursion, including metal-water reactions, is approximately 18 megajoules. This energy release is equivalent to the shock energy released by the explosion of 38 lbs of TNT. Lower excursion energies will have a correspondingly lower TNT equivalent. At excursion energies of less than 100 megajoules, where the pressure tubes are not explosively ruptured, the release of shock energy is negligible.

Extensive fragmentation of metal parts may occur as a result of jetting. The jetting occurs at the intersection of the cylindrically expanding shock waves from adjoining exploding fuel elements. The fragment velocity may be several times the material velocity in these shocks, or roughly 5000 ft/sec.

If the energy of the excursion is compared to an equal energy from TNT* where W = lb TNT and R = distance in ft.

$$\text{Peak pressure, } P_m = 1.4 \times 10^4 \frac{W^{1/3}}{R} \text{ lb/in}^2$$

$$\text{Impulse} \qquad I = 1.9 \frac{W^{2/3}}{R} \text{ lb sec/in}^2$$

$$\text{Energy per area, } \Omega = 195 \frac{W}{R^2} \text{ ft lb/in}^2$$

$$\text{Maximum particle velocity at the front } v_m = 230 \frac{W^{1/3}}{R} \text{ ft/sec}$$

$$\text{Characteristic time, } \frac{1}{n} = \frac{W^{1/3}}{9150} \text{ sec, where } P = P_m e^{-nt}$$

At the pool wall R = 10, and when W = 38, then

$$P_m = 4700 \text{ lb/in}^2$$

$$I = 2.1 \text{ lb sec/in}^2$$

*Cole, R. H. Underwater Explosions, Princeton University Press

Ω = 74 ft lb/in^2

v_m = 77 ft/sec

$\frac{1}{n}$ = 0.37 msec

at the surface of the pool R = 25 and

P_m = 750 lb/in^2

These values may be taken as pessimistic. The reactor core is larger in volume than the volume of 38 lb of TNT. The release of energy from an excursion requires a longer time than the release of energy in a TNT explosion. Thus, the specific energy density and the resultant peak overpressures are lower in the core than in TNT. The maximum pressure in the vaporized center of the core was found to be 24000 psi (see App. A), while the equivalent energy release by TNT would result in about 47000 psi at a radius of 1 ft. In addition, the reactor structure and 4-in.-thick lead shield will absorb considerable shock energy and further reduce the overpressures.

Thus for the same energy (Ω), the width of the shock wave resulting from an exploding reactor will be substantially larger than that from a charge of TNT, while the pressure will be less.

Using the equivalent TNT pressure as the maximum condition, the pressure that will be exerted on the pool wall is 4700 psi. Local failure in the concrete walls may occur as the result of shock reflections or from metal fragments. A shock impulse of 0.8 lb sec/in^2 could cause the concrete to crack but no gross structural failure of pool walls will occur.

5. Pressure in the Reactor Room

As the pressure wave passes from the water into the air, the pulse transmitted at the interface has a very small overpressure compared to the overpressure in the water. The velocity potential (ϕ_1) of the refracted wave is given in terms of that of the incident wave (ϕ) by*

$$\frac{\phi_1}{\phi} = \frac{2}{\frac{\rho_1}{\rho} + \frac{\cot \theta_1}{\cot \theta}}$$

where ρ is the density and θ is the incident angle for the incident medium and ρ_1 and θ_1 are the density and incident angle for the refractive medium.

$$\frac{\cot \theta_1}{\cot \theta} = \frac{v}{v_1} \sqrt{1 - \left(\frac{v_1^2}{v^2} - 1\right) \tan^2 \theta}$$

* Rayleigh, Baron, Theory of Sound, England, Dover Press, 1878

$$= \frac{v}{v_1} = 4.3 \text{ for normal incidence}$$

and $$\frac{\rho_1}{\rho} = \frac{1}{800}$$

so that $$\frac{\phi_1}{\phi} = 0.46$$

The overpressure is proportional to ϕ, so that at normal incidence

$$\frac{P_1}{P} = \frac{\rho_1}{\rho}\frac{\phi_1}{\phi} = \frac{0.46}{800} \approx 6 \times 10^{-4}$$

For the pessimistic assumption that the overpressure is equivalent to the detonation of TNT with equal energy, the shock transmitted into the air, at the surface of the pool is

$$P_{air} = (750)\ (6 \times 10^{-4}) = 0.45 \text{ psi}$$

Essentially no physical damage is expected from this overpressure.

When the bubble of compressed gasses, resulting from the excursion, rises to the surface, the resultant "geyser" will eject large quantities of water from the pool. It is unlikely that the massive pieces of metal will be carried up through 20 ft of water. The reactor building will remain virtually intact, although local leaks may occur.

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APPENDIX C

FISSION PRODUCT DISTRIBUTION FOR MAXIMUM ACCIDENT

A. INTRODUCTION

The following assumptions were made in calculating the radiation effects that would result from a maximum credible accident:

1. The reactor has been operating at 2 megawatts for 10^4 hr prior to the excursion.

2. The integrated excursion energy was 400 megajoules; (it was calculated to be 380 megajoules). This assumption is not critical because the dosage contribution from steady state accumulation of fission products is greater than the contribution from the excursion except in the reactor building.

3. Ten percent of the total fission product activity escapes from the reactor vessel, up through the pool, and is released into the reactor room.

4. The fission products activity is released into the atmosphere in the form of a cloud with a radius of 20 meters, located 25 meters above ground level.

5. The diffusion theory of O. G. Sutton is valid in calculations of down-wind cloud concentration. The following values of the atmosphere diffusion coefficient were used for specific values of wind velocity, u, in meters/sec. and stability parameter, n.

	u = 1	u = 2	u = 5	u = 10	u = 20
n = 0.2	.30	.28	.25	.23	.21
n = 0.5	.08	.07	.05	.04	.04

B. DIRECT RADIATION

The direct gamma dosages were calculated from AECU 3066 using the nomographs constructed by J. Z. Holland. The resultant dosages are shown in Figure C-1 and selected values are given in the table on page C-7.

C. DOSE RATE FROM FALL-OUT AND RAIN-OUT

The method used in calculating the surface deposition of radioactive products is that developed in KAPL-1045. The fundamental assumption is that a constant fraction of the radioactivity in the cloud falls out per unit

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distance traveled by the cloud. For each distance, this fall-out rate is taken to be that which would give the maximum dose at that point. This method over-estimates the dose at all distances. The ground fallout concentration is shown to be

$$\frac{C}{er} \quad \frac{e^{-y^2/\sigma^2}}{\pi \sigma^2}$$

where

C = total cloud activity in curies, neglecting previous fallout
= $1.72 \times 10^9 \ t^{-1.2} + 4.32 \times 10^7 \ t^{-0.2}$

σ = cloud radius

r = downwind distance

y = horizontal distance from cloud path

Neglecting the correction for the Gaussian distribution of cloud activity (which would reduce the resultant value) but correcting for air attenuation ($\lambda \equiv 200$ meters, attenuation length), the dose rate at a distance h meters above the ground is:

$$\frac{2 \ \pi \quad (0.57)C}{\sigma} \int_h^\infty \frac{e^{-t}}{t} \, dt$$

The resulting values are shown in Figure C-2 and selected values are given in the table on the following page. Neglecting the non-uniform cloud distribution will result in an over estimate of dose by a factor of 2, at the distances where the cloud radius is still about 20 meters.

D. INHALATION DOSE

The internal dose received due to inhalation of a particular isotope is given by the equation

$dA = f_o \ J \ C'(r) \ dt$

A = dose in curies

f_o = body retention of inhaled material

J = inhalation rate = $17 \frac{\text{liters}}{\text{min}} = .284 \frac{\text{liters}}{\text{sec}}$

C'(r) = cloud concentration at downwind distance r

For a Gaussian distribution,

$$C'(r) = \frac{Q}{V} \ e^{-(r'/\sigma)^2}$$

where Q = quantity of isotope contained in cloud

$V = \text{cloud volume} \approx (\sqrt{\pi}\sigma)^3$

r' = distance from cloud center

TYPICAL VALUES OF IRRADIATION FOR CLOUD CONTAINING 10% OF FISSION PRODUCTS

2-MEGAWATT STEADY STATE AND 400 MEGAJOULE EXCURSION

n	WIND VELOCITY M/SEC	DOWNWIND DISTANCE M	CLOUD RADIUS M	AVERAGE TIME SEC	WASH-OUT RADIATION EXCURSION PRODUCTS R/HR	WASH-OUT RADIATION STEADY-STATE PRODUCTS R/HR	WASH-OUT RADIATION TOTAL R/HR	DIRECT CLOUD RADIATION EXCURSION PRODUCTS R	DIRECT CLOUD RADIATION STEADY-STATE PRODUCTS R	DIRECT CLOUD RADIATION TOTAL R
0.2 (Large Lapse)	1	10^2	25	10^2	100	260	360	5.5	12	17.5
		10^3	160	10^3	0.1	2.5	2.6	0.06	1.6	1.7
		10^4	1100	10^4	0.0001	0.023	0.023	0.001	0.024	0.024
	10	10^2	23	10	1800	450	2300	6.5	1.9	8.4
		10^3	140	10^2	1.8	4.5	6.3	0.16	0.4	0.56
		10^4	900	10^3	0.018	0.044	0.046	0.0003	0.006	0.006
0.5 (Large Inversion)	1	10^2	20	10^2	130	320	450	5.0	11.0	16.0
		10^3	30	10^3	0.54	13.5	14	0.25	6.8	7.0
		10^4	90	10^4	Neg.	0.29	0.29	0.006	1.5	1.5
		10^5	450	10^5	Neg.	0.0036	0.0036	Neg.	0.080	0.080
	10	10^2	20	10	2000	500	2500	6.5	1.9	8.4
		10^3	20	10^2	11	27	38	0.40	1.0	1.4
		10^4	23	10^3	0.0034	0.85	0.88	0.02	0.6	0.62
		10^5	47	10^4	Neg.	0.012	0.012	Neg.	0.046	0.046

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and noting that $dt = \frac{dr'}{u}$, then

$$A = f_o \frac{J}{u} \frac{Q}{V} \int_{-\infty}^{+\infty} e^{-(r'/\sigma)^2} dr'$$

$$= \frac{f_o J Q}{u \pi \sigma^2}$$

For any particular isotope,

$$Q = f e^{-\lambda r/u} [F (1-e^{-\lambda T}) + G \lambda]$$

where

f = fractional fission yield of isotope

λ = isotope decay constant

F = average fission rate of reactor = 6×10^{16}/sec

G = reactor operating time = 10^4 hours

Appropriate values of f, f_o and λ are given in the table on page C-7 for the important isotopes. The decay constant of these isotopes is small, so that the decay is essentially negligible out to distances of 10^5 meters from the origin. These six isotopes contribute approximately 85% of the total body inhalation dose.

The resulting values are plotted in Figures C-3 and C-4 for microcuries deposited in the body and in Figures C-5 and C-6 for infinite time dosages.

E. DOSAGE RECEIVED BY PERSONNEL IN THE CONTROL ROOM

The Wigner-Way relation for gamma power from fission products can be expressed as

$$1.28\ t^{-1.2} \ \frac{\text{mev}}{\text{sec fission}} \text{ where t is in sec.}$$

For 400 megajoule excursion

$$(4 \times 10^8)\ (3.1 \times 10^{10})\ (1.28\ t^{-1.2}) = 1.59 \times 10^{19}\ t^{-1.2} \text{ mev.}$$

The energy released in the first ten minutes (or 600 sec) is

$$1.59 \times 10^{19} \int_1^{600} t^{-1.2}\, dt = \frac{1.59 \times 10^{19}}{-0.2} \left[t^{-0.2}\right]_1^{600}$$

$$= \frac{(1.59 \times 10^{19})}{-0.2} (.278-1) = 5.73 \times 10^{19} \text{ mev.}$$

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For long term operation at steady state the Wigner-Way relation for gamma power from build-up fission products can be expressed as

$$1.97 \times 10^{11} \; P \; t^{-0.2} \text{ mev/sec},$$

where P is in watts and t is sec after shutdown. At 2 Mw steady state operation $(2 \times 10^6)\ (1.97 \times 10^{11})\ t^{-0.2} = 3.94 \times 10^{17}\ t^{-0.2}$ mev/sec.

Energy release in first 10 minutes (or 600 sec) is

$$3.94 \times 10^{17} \int_1^{600} t^{-0.2} \; dt = \frac{3.94 \times 10^{17}}{0.8} \left[t^{0.8} \right]_1^{600}$$

$$= \frac{(3.94 \times 10^{17})}{0.8} \; (167) = 8.23 \times 10^{19} \text{ mev.}$$

The total gamma source in first 600 sec is

$$5.73 \times 10^{19} + 8.23 \times 10^{19} = 1.4 \times 10^{20} \text{ mev.}$$

Only 10% of the fission products reach the pool surface, resulting in a source of

$$(.1)\ (1.4 \times 10^{20}) = 1.4 \times 10^{19} \text{ mev.}$$

Letting,

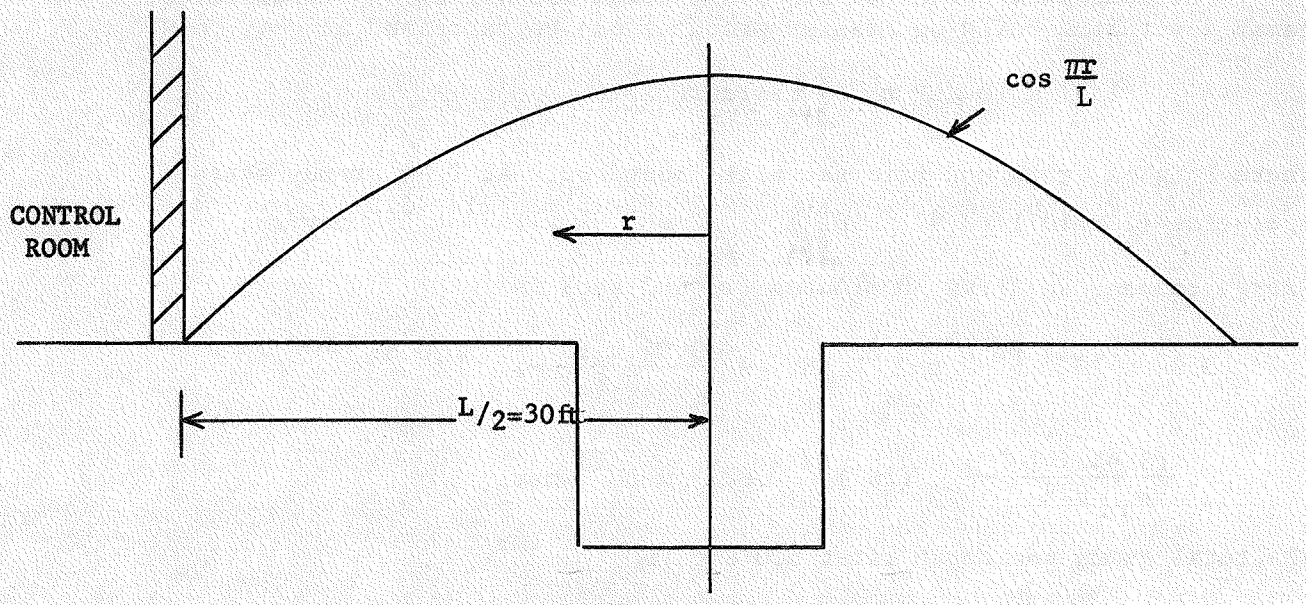
$$5.5 \times 10^5 \; \frac{\text{mev}}{\text{cm}^2 \text{ sec}} = 1 \text{ R/hr over energy range of interest.}$$

Then, the source during the initial 10 minutes is

$$\frac{1.4 \times 10^{19} \text{ mev}}{(5.5 \times 10^5)\ (600 \text{ sec})} \; (1/6 \text{ hr}) = 7.04 \times 10^9 \text{ cm}^2 \text{ R.}$$

The fission products released into the reactor room will be in two forms: gas, which will tend to diffuse into the entire room, and small solid fragments or dust dissolved in the water, both of which will be blown into the air and will fall with their majority concentrated near the pool. It is assumed that this distribution of fission products may be approximated by a one dimension cosine distribution centered at the center of the pool and going to zero at the concrete wall between the reactor room and the control room.

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$$\text{Source } S = A \cos \frac{\pi r}{L} = \frac{\pi Q}{2L} \cos \frac{\pi r}{L}$$

where Q = total dose = 7×10^9 cm^2 R.

$$\text{Dosage } D = \int_{L/2}^{-L/2} \frac{S\,dr}{4\pi r^2} = \int_{L/2}^{-L/2} \frac{(7 \times 10^9)\,\pi}{(2L)\,(4\pi)} \cos\left(\frac{\pi r}{L}\right) \frac{dr}{r^2}$$

$$D = 2260R$$

$$\text{Attenuation due to 1.5 foot concrete wall } e^{-\frac{(1.5)\,(30.5)}{15}} \approx \frac{1}{20}$$

$$\therefore \frac{2260}{20} = 113R.$$

Thus the dosage at the wall inside the control room for a 10 minute period is 113R. It should be noted that approximately one half of the total dosage is received in the first 6 seconds.

Some additional inhalation dose might be received due to inhalation of fission product gases (krypton and xenon) which would not be removed by air filtration. These products are not deposited in the body.

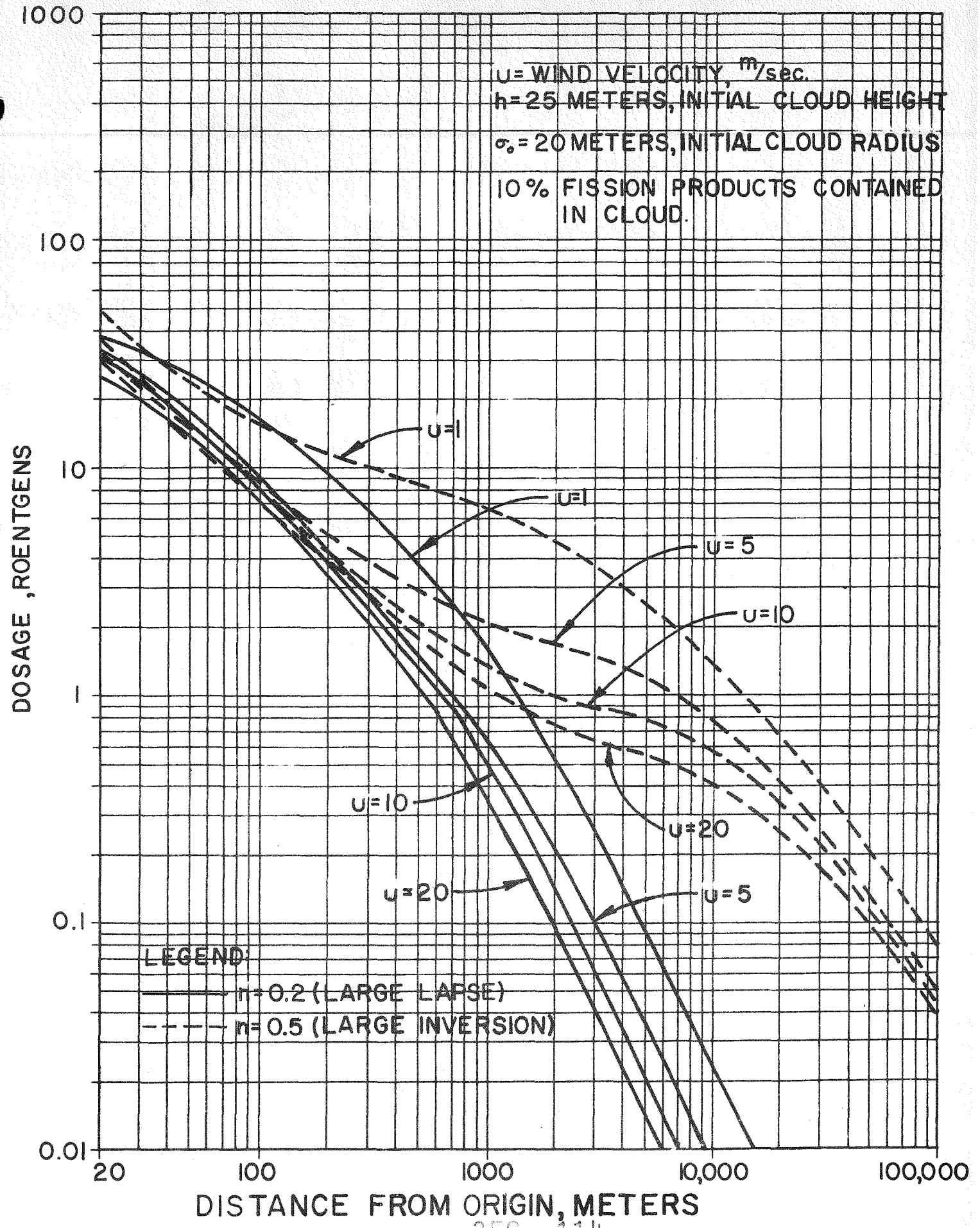
The evacuation procedure has been modified such that the counting room (which is 50 feet further removed from the pool) is used as a shelter, rather than the control room. Thus the dosage received by personnel in the counting room will be approximately 18R (using r^2 attenuation of the dose).

DATA FOR IMPORTANT FISSION ISOTOPES

Isotope	Critical Organ	Inhalation Uptake	Fission Yield	Half Life Days	Permissible Body Burden Microcuries
I^{131}	Thyroid	0.15	.028	8.1	0.3
Sr^{89}	Bones	0.22	.046	51	2
Y^{91}	Bones	0.14	.059	58	15
$Ba^{140}+La^{140}$	Bones	0.2	.062	12.8	5
$Ce^{144}+Pr^{144}$	Bones	0.1	.053	285	5
$Sr^{90}+Y^{90}$	Bones	0.22	.059	10^4	1

These six isotopes contribute over 85% of the total body inhalation dose.

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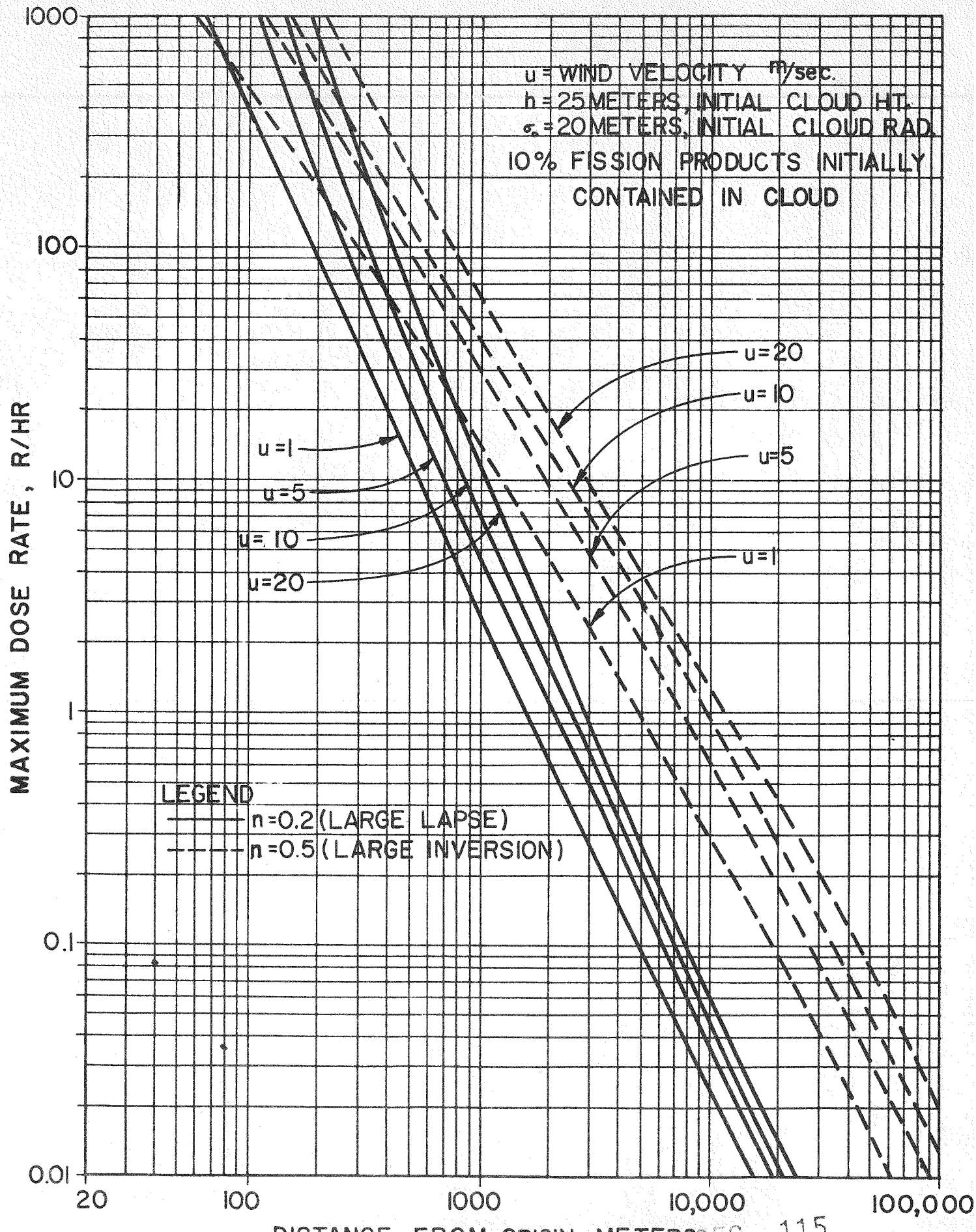
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FIG. C-1

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Embedded Image

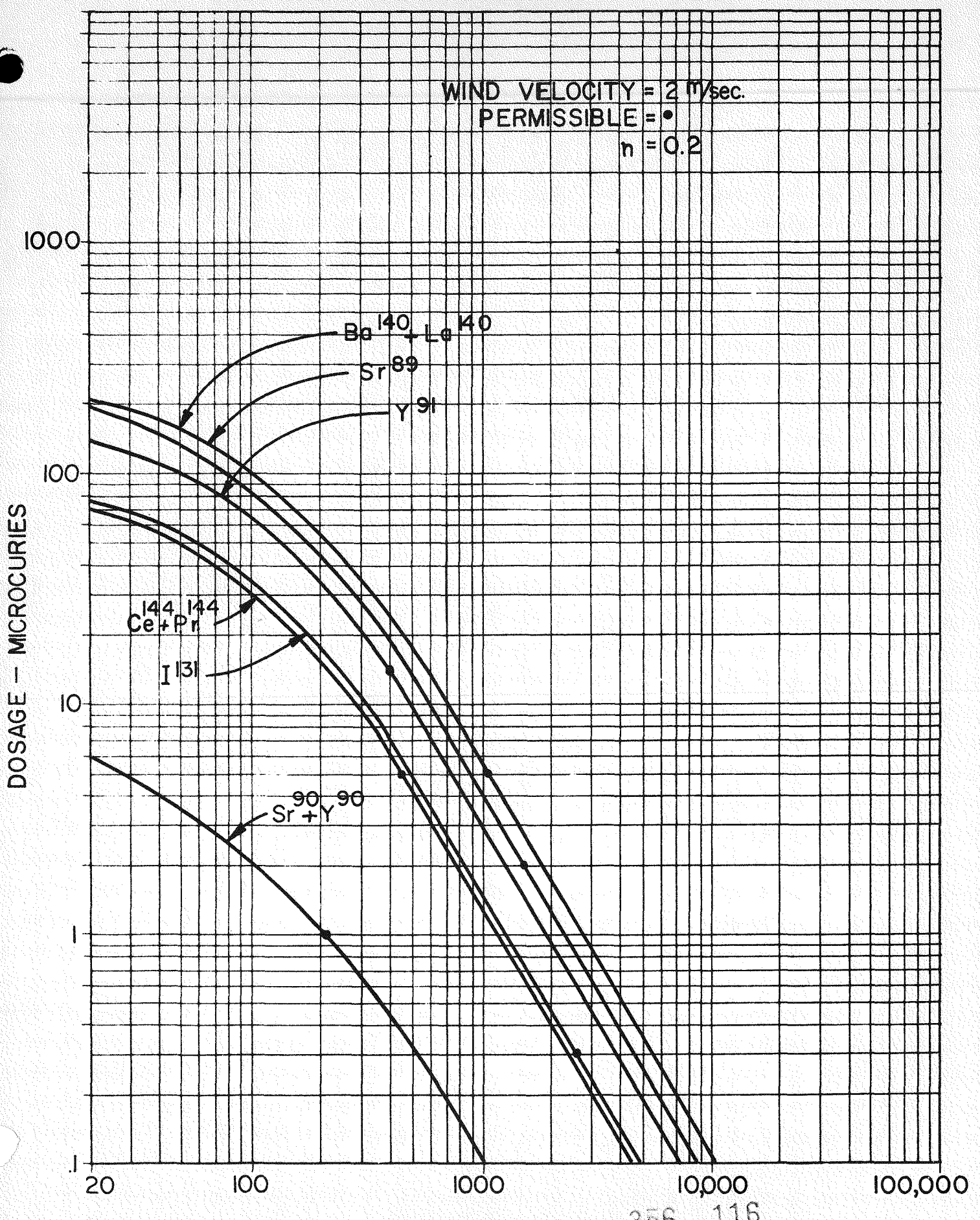
RADIATION DUE TO CLOUD WASHOUT



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FIG.-C-2

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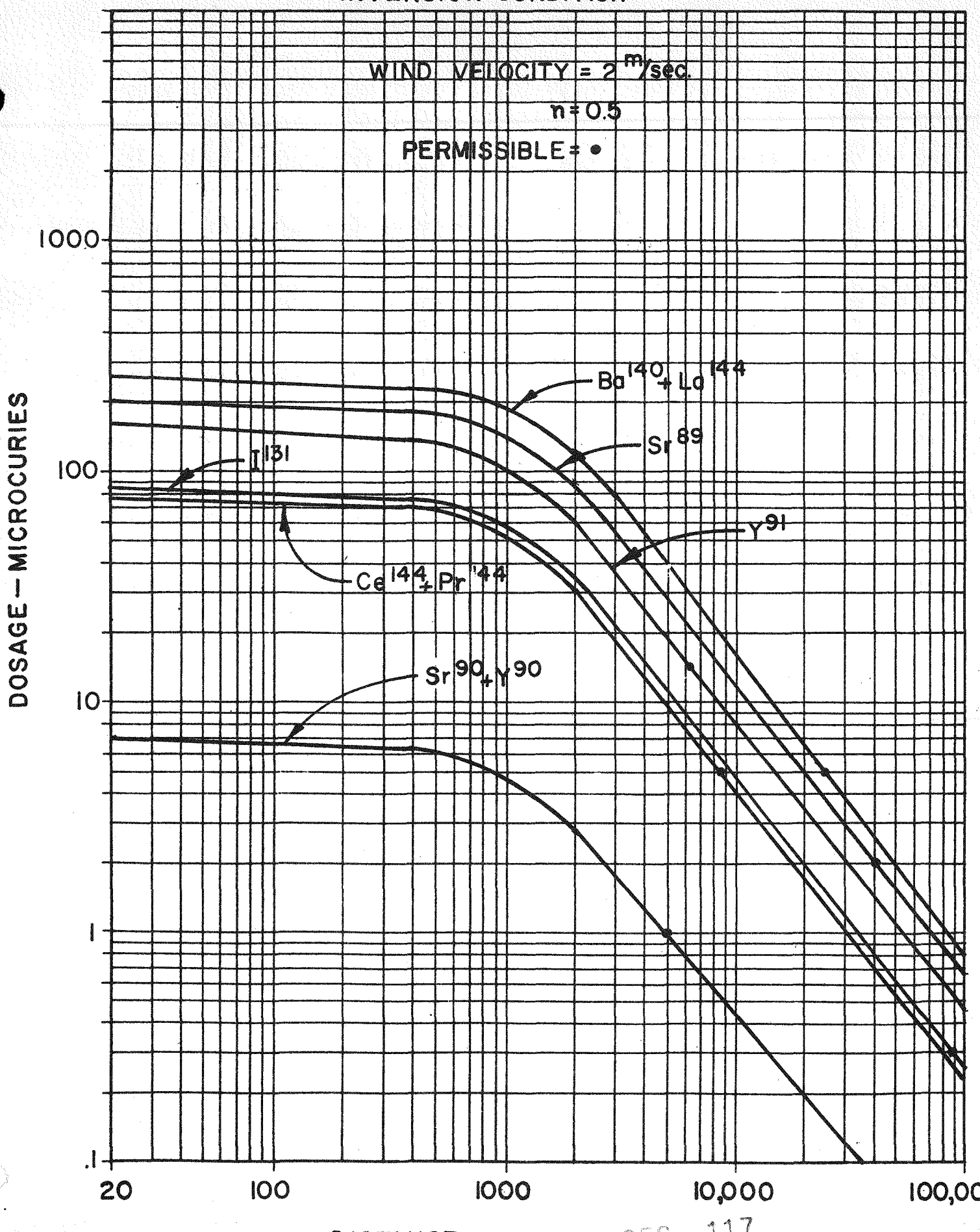


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FIG.-C-3

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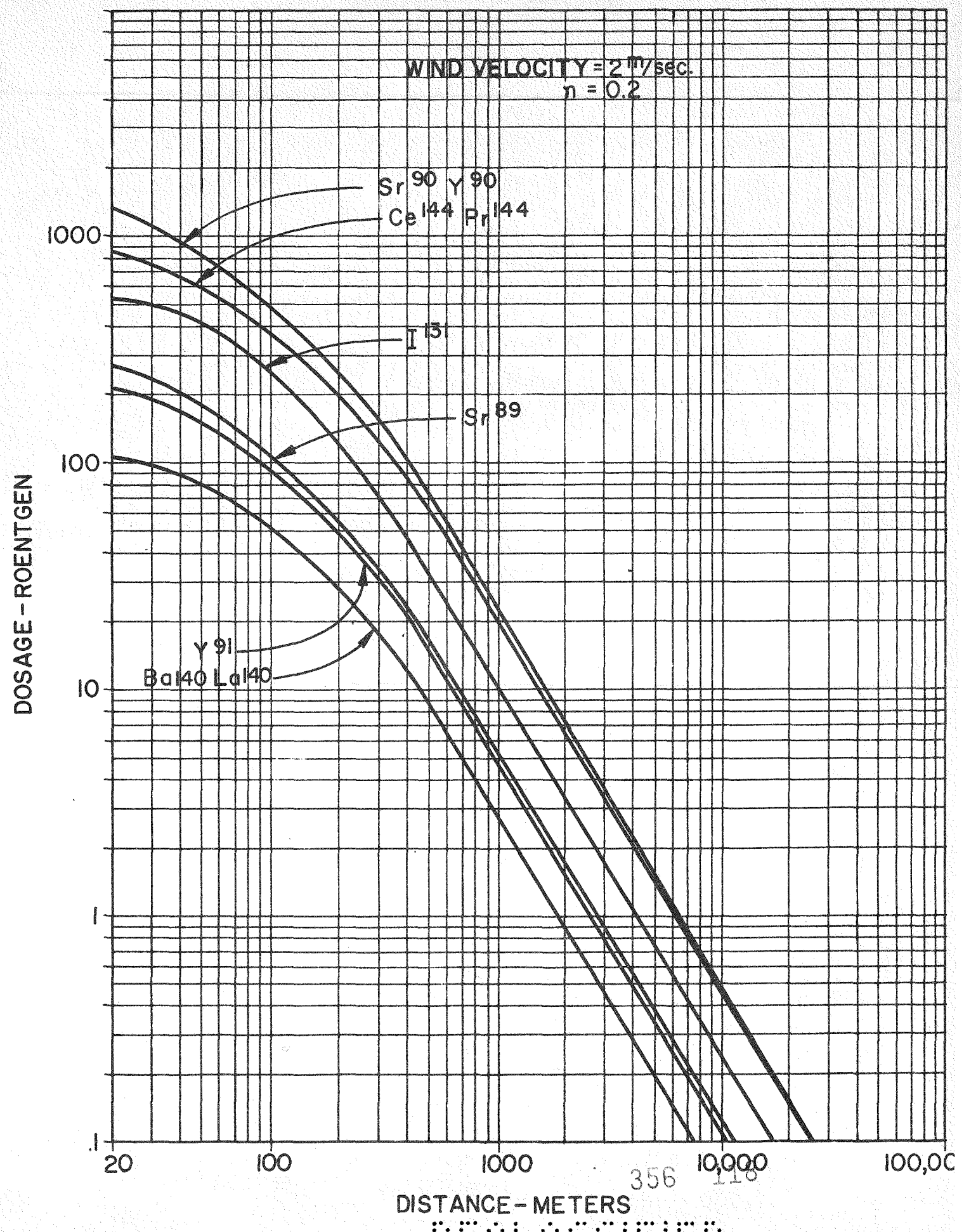
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FIG-C-4

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FIG-C-5

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INFINITE TIME DOSAGE
FOR INVERSION CONDITION

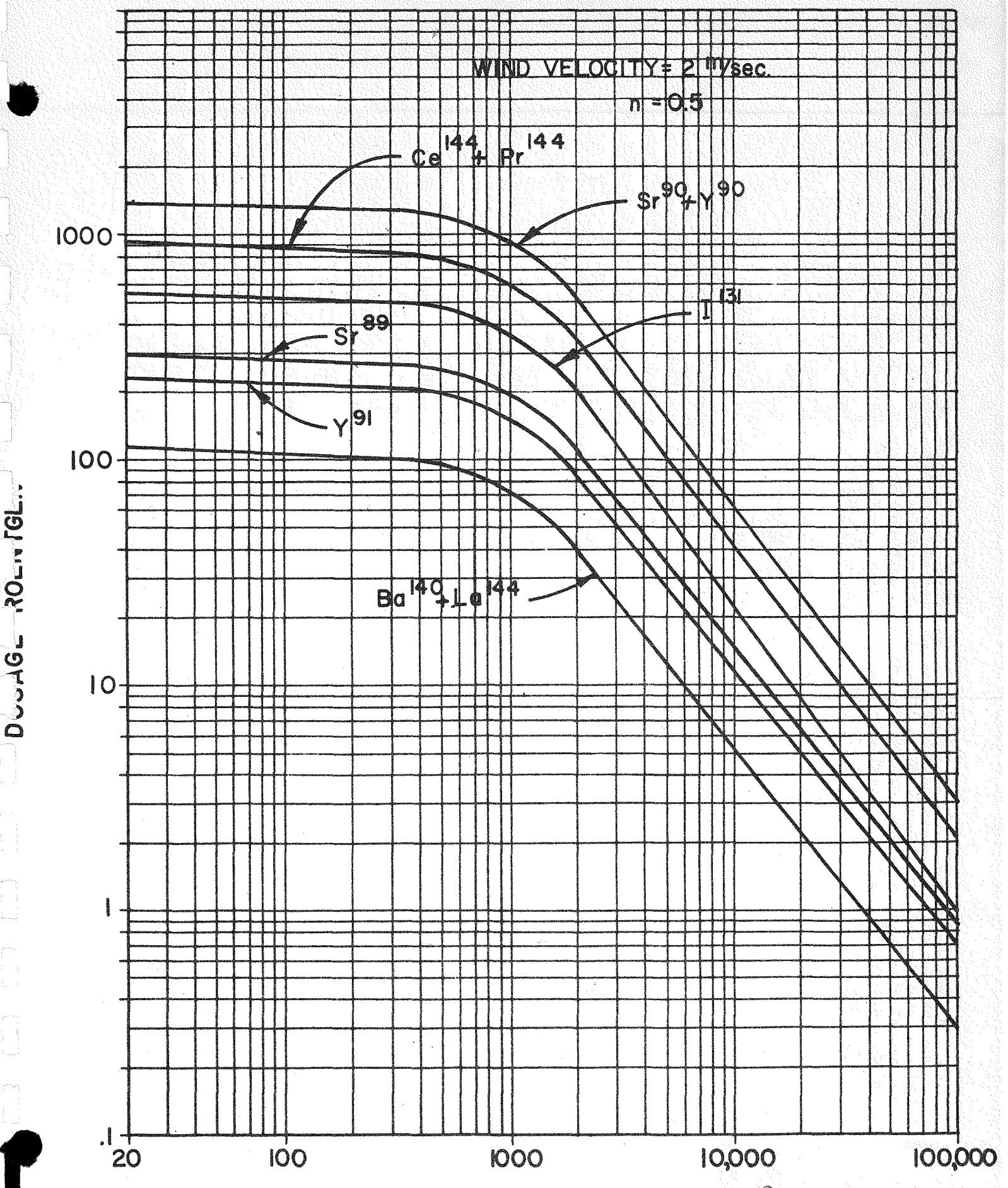


FIG.-C-6

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APPENDIX D

U. S. DEPARTMENT OF COMMERCE
WEATHER BUREAU

LOCAL CLIMATOLOGICAL DATA

WITH COMPARATIVE DATA

1956

IDAHO FALLS 46 W, IDAHO

NARRATIVE CLIMATOLOGICAL SUMMARY

The National Reactor Testing Station (NRTS) covers about 680 square miles of the upper Snake River Plain. The most important topographic features are the northeast-southwest orientation of the Plain, and the Continental Divide to the north and east. About 90 percent of the time surface winds are either southwesterly or northeasterly. The Divide prevents the invasion of many of the cold continental air masses in winter. All air masses that enter the Plain must first cross a mountain barrier, regardless of the direction from which they enter. Because they are subject to lifting, these masses usually precipitate moisture over the mountains, and enter the Plain sufficiently dry to give the region its desert-like characteristics.

An important local topographic effect is due to the Lemhi Range of mountains just west of the NRTS. The winds aloft are mostly westerly and this places the NRTS in the immediate lee of the mountains, in a region of descending air currents. When contrasted with other locations on the Snake River Plain that are exposed to more direct flow, (such as Pocatello or Idaho Falls), the drying of the air resulting from the descending motion is evident. A few obvious consequences are:

1. The annual precipitation totals on the NRTS are only about half as much as Idaho Falls and Pocatello.
2. Clear skies are frequently observed over the NRTS when it is cloudy elsewhere over the Plain.
3. The depth of snow on the ground is less at the NRTS than at Idaho Falls.
4. Cloudiness and precipitation from major storms often begin sooner and last longer at Idaho Falls, 45 miles to the east, than at the NRTS.

The dryness and the infrequency of low cloudiness permit intense solar heating of the ground during the day, and rapid radiational cooling at night, which results in a very large daily range of temperature. The average range is greatest (about 37°F.) in summer and least (about 23°F.) in winter. Summers are especially pleasant because of the dryness and the cool nights. The dryness also makes more bearable the extreme cold spells in winter, as does the fact that these cold periods are frequently accompanied by clear skies and are seldom associated with strong winds.

Temperature inversions based at the surface are present for at least a few hours almost every day of the year and with snow cover may persist for several consecutive days. Temperature inversions through the lower 250 feet of the atmosphere are present for about half of the total time for the year as a whole. While temperature measurements have not been made to the top of the inversions at the times of maximum depths, the average maximum depth is estimated to be approximately 2,000 feet.

There is a strong tendency for up-valley flow (southwest wind) during the day and down-valley flow (northeast wind) at night. The local effect is sometimes countered and sometimes augmented by the pressure gradient, or the larger scale features of the wind flow pattern. Frequently the local effects dominate the wind flow near the ground in the face of strongly opposing pressure forces. On many occasions the station, Idaho Falls 42 NW, will experience northerly winds while the station, Idaho Falls 46 W, has southerly winds. This situation arises at times from drainage effects and at other times from a topographically induced eddy.

The heaviest precipitation falls as scattered rain showers during the warmer months, and the rain from a single passing shower may in a matter of minutes exceed considerably the normal total for the month in which it occurs. Yet two or three months may pass in summer without a single shower passing the station. Despite this extreme variability, annual totals over the NRTS have not varied greatly from year to year or from place to place. One inch of rainfall in an hour is about the maximum intensity to be expected, and two inches in 24 hours are not likely to be exceeded. Maximum winter snow depth has been extremely variable

(Continued on last page)

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LATITUDE 43° 32' N
LONGITUDE 112° 57' W
ELEVATION (ground) 4933 FEET

METEOROLOGICAL DATA FOR THE CURRENT YEAR

IDAHO FALLS, 46 W, [redacted] 1956

Month	Temperature: Averages: Daily maximum	Daily minimum	Monthly	Extremes: Highest	Date	Lowest	Date	Degree days	Precipitation: Total	Greatest in 24 hrs.	Date	Snow, Sleet: Total	Greatest in 24 hrs.	Date	Relative humidity				Wind: Average hourly speed	Prevailing direction	MAXIMUM HOURLY AVERAGE: Speed	Direction	Date	Percent of possible sunshine	Average sky cover sunrise to sunset	Number of days: Sunrise to sunset: Clear	Partly cloudy	Cloudy	Precipitation .01 inch or more	Snow, Sleet 1.0 inch or more	Thunderstorms	Heavy fog	Temperatures: Maximum: 90° and above	32° and below	Minimum: 32° and below	Zero and below
JAN.	32.2	13.0	22.6	46	7	-30	31	1309	0.80	0.18	25-26	10.5	4.4	25-26					6.2	WSW	39	WSW	8						15	2	0		0	13	31	5
FEB.	26.8	1.4	12.7	50	22	-29	2#	1514	0.31	0.20	19-20	4.8	3.7	19-20					7.8	WSW	33	WSW	23						7	1	0		0	20	29	17
MAR.	41.9	16.1	29.0	63	24	-10	11	1111	0.27	0.09	10#	4.1	2.0	4					10.7	WSW	41	WSW	26						4	2	0		0	8	30	6
APR.	59.1	29.1	44.1	76	21	15	6	619	0.35	0.13	23-24	T	T	29#					8.5	WSW	30	WSW	26#						8	0	2		0	0	19	0
MAY	70.0	40.4	55.2	87	31	24	1	302	1.63	0.62	28	T	T	12					8.8	WSW	31	WSW	10						9	0	4		0	0	5	0
JUN.	78.7	42.6	60.7	93	28	30	22#	145	0.02	0.01	23#	0.0	0.0						10.9	WSW	34	WSW	4						2	0	5		3	0	2	0
JUL.	87.9	51.0	69.5	96	10#	37	1	21	0.17	0.11	28-29	0.0	0.0						7.3	WSW	29	WSW	6						4	0	3		15	0	0	0
AUG.	83.6	44.5	64.1	95	23#	30	31	83	0.02	0.01	26#	0.0	0.0						7.3	WSW	31	WSW	29						2	0	4		6	0	1	0
SEP.	77.4	36.5	57.0	89	19#	22	23#	235	0.28	0.19	11-12	0.0	0.0						8.1	WSW	33	SW	30						4	0	1		0	0	7	0
OCT.	58.6	27.4	43.0	77	7	12	29	673	1.53	0.74	27	2.4	1.6	30-31					8.8	WSW	32	WSW	21						9	1	0		0	0	24	0
NOV.	39.7	11.4	25.6	57	12	- 2	20	1178	0.28	0.13	13	3.1	1.8	4					3.8	NNE	30	SW	13						3	1	0		0	7	30	2
DEC.	32.9	8.0	20.5	47	4	-13	8	1375	0.15	0.06	10-11	1.5	0.6	10-11					4.7	NNE	30	WSW	6						5	0	0		0	16	31	7
Year	57.4	26.6	42.0	96	JUL. 10#	-30	JAN. 31	8565	5.81	0.74	OCT. 27	26.4	4.4	JAN. 25-26					7.7	WSW	41	WSW	MAR. 26						72	7	19		24	64	209	37

NORMALS, MEANS, AND EXTREMES

Month	Temperature: Normal: Daily maximum	Daily minimum	Monthly	Extremes: Record highest	Year	Record lowest	Year	Normal degree days	Precipitation: Normal total	Maximum monthly	Year	Minimum monthly	Year	Maximum in 24 hrs.	Year	Snow, Sleet: Mean total	Maximum monthly	Year	Maximum in 24 hrs.	Year	Relative humidity				Wind: Mean hourly speed	Prevailing direction	MAXIMUM HOURLY AVERAGE: Speed	Direction	Year	Pct. of possible sunshine	Mean sky cover sunrise to sunset	Mean number of days: Sunrise to sunset: Clear	Partly cloudy	Cloudy	Precipitation .01 inch or more	Snow, Sleet 1.0 inch or more	Thunderstorms	Heavy fog	Temperatures: Max.: 90° and above	32° and below	Min.: 32° and below	0° and below
(a)	(b)	(b)	(b)	7		7		(b)	(b)	7		7		7		7	7		7						6	6	6	6						7	7	7		7	7	7	7
J	27.6	3.8	15.7	49	1953	-30	1956	1528	0.66	0.96	1950	0.45	1953	0.40	1954	6.3	10.5	1956	4.4	1956					7.2	SW	39	WSW	1956						10	3	0		0	20	31	10
F	32.9	8.2	20.6	53	1953	-29	1956	1243	0.58	1.04	1952	0.21	1955	0.54	1954	4.8	10.0	1952	6.5	1950					7.4	WSW	26	SW	1955						6	1	0		0	13	28	9
M	41.9	18.0	30.0	66	1953	-15	1955	1085	0.45	1.05	1950	0.20	1955	0.30	1950	4.5	7.5	1950	5.8	1953					9.7	SW	51	WSW	1955						6	2	*		0	8	31	4
A	56.0	29.0	42.5	77	1952	6	1953	675	0.75	0.64	1953	0.03	1952	0.34	1953	0.7	3.8	1955	3.0	1955					9.1	SW	39	WSW	1954						5	*	1		0	0	22	0
M	67.1	38.0	52.6	91	1954	18	1954	390	1.02	1.78	1953	0.28	1950	0.70	1953	0.7	4.7	1955	3.1	1955					9.2	SW	37	W	1954						7	*	2		*	0	10	0
J	75.5	43.6	59.6	95	1955#	26	1954	172	0.81	2.74	1954	0.02	1956	1.73	1954	T	T	1954	T	1954					9.4	SW	35	WSW	1954						6	0	2		2	0	3	0
J	88.1	50.3	69.2	99	1955#	33	1954	16	0.44	0.65	1955	0.00	1953	0.29	1955	0.0	0.0		0.0						8.0	WSW	35	WSW	1955						3	0	2		13	0	0	0
A	85.7	48.1	66.9	99	1954	28	1954	71	0.64	2.17	1951	0.02	1956	0.72	1950	0.0	0.0		0.0						7.5	WSW	31	WSW	1956						4	0	4		8	0	*	0
S	73.4	38.6	56.0	96	1955	18	1954	286	0.36	0.28	1956#	T	1951	0.19	1956	T	0.1	1950	0.1	1950					7.3	SW	38	WSW	1955						2	0	1		2	0	7	0
O	60.5	27.3	43.9	82	1952	9	1954	654	0.68	1.53	1956	0.00	1952	0.74	1956	0.4	2.4	1956	1.6	1956					6.6	SW	44	WSW	1954						4	*	*		0	0	24	0
N	42.9	17.0	30.0	63	1954#	-24	1955	1050	0.56	0.65	1954	0.08	1953	0.51	1954	1.3	3.1	1956	1.8	1956					5.8	SW	40	WSW	1955						4	1	0		0	5	28	3
D	31.4	9.1	20.3	50	1955	-18	1952	1386	0.75	2.06	1955	0.15	1956	0.52	1955	4.6	7.2	1951	3.0	1950					6.1	WSW	43	SW	1955						8	2	*		0	17	31	9
Year	56.9	27.6	42.3	99	JUL. 1955#	-30	JAN. 1956	8556	7.70	2.74	JUN. 1954	0.00	JUL. 1953#	1.73	JUN. 1954	23.3	10.5	JAN. 1956	6.5	FEB. 1950					7.8	SW	51	WSW	MAR. 1955						65	9	12		26	63	215	35

(a) Length of record, years. (b) Normal values are based on the period 1921-1950, and are means adjusted to represent observations taken at the present standard location
OCCURRENCES BETWEEN 5:00 P.M. AND 8:00 A.M. REPORTED BY UNOFFICIAL OBSERVERS.
LEAP YEAR FEBRUARY AND ANNUAL PRECIPITATION NORMALS ARE INCREASED BY NORMAL PRECIPITATION FOR FEBRUARY 29.

AVERAGE TEMPERATURE

Year	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	Ann'l
1950	13.6	21.2	30.6	40.0	48.2	57.2	66.1	64.9	54.9	48.1	32.2	25.5	41.9
1951	13.1	23.2	27.4	43.8	53.5	56.2	68.9	64.8	56.2	41.6	28.5	15.5	41.1
1952	11.8	19.2	20.2	42.8	54.0	59.5	66.3	66.2	58.2	46.0	23.9	18.8	40.0
1953	30.0	26.8	33.4	39.8	45.4	57.0	69.7	65.7	59.1	46.0	35.0	17.6	43.8
1954	23.8	29.8	29.7	44.0	53.6	56.2	69.9	63.8	54.6	41.2	35.1	16.6	43.2
1955	8 4	12.6	23.7	37.8	49.8	59.0	68.3	68.5	55.1	43.9	25.9	24.3	39.8
1956	22 6	12.7	29.0	44.1	55.2	60.7	69.5	64.1	57.0	43.0	25.6	20.5	42.0
RECORD MEAN													
TEMP	17.6	19.5	27.8	41.8	51.4	58.0	68.4	65.5	56.5	44.6	29.5	19.9	41.7
MAX	28.2	32.3	39.4	56.5	66.8	74.5	87.1	84.6	75.8	62.1	43.2	30.9	56.8
MIN	6.9	6.6	16.1	27.0	35.9	41.5	49.6	46.3	37.1	27.0	15.7	8.8	26.5

TOTAL PRECIPITATION

IDAHO FALLS, 46 W, IDA. 1956

Year	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	Annual
1950	.96	.75	1.05	.14	.28	.98	.10	1.08	.28	.16	.30	.55	6.63
1951	.76	.58	.60	.30	.56	.15	.58	2.17	T	.69	.33	.72	7.44
1952	.61	1.04	.41	.03	.80	.91	.38	.29	.12	.00	.37	.67	5.63
1953	.45	.25	.40	.64	1.78	.80	.00	.18	.01	.41	.08	.25	5.25
1954	.51	.87	.55	.46	.45	2.74	.16	.39	.15	.07	.65	.40	7.40
1955	.47	.21	.20	.49	.81	1.03	.65	.19	.21	.20	.24	2.06	6.76
1956	.80	.31	.27	.35	1.63	.02	.17	.02	.28	1.53	.28	.15	5.81
RECORD MEAN	.65	.57	.50	.34	.90	.95	.29	.62	.15	.44	.32	.69	6.42

MONTHLY AND SEASONAL DEGREE DAYS

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1949-50	-	-	-	-	-	-	1593	1228	1066	750	524	234	-
1950-51	38	48	340	524	959	1225	1617	1170	1169	637	370	274	8371
1951-52	31	76	263	733	1094	1553	1649	1589	1389	666	341	165	9549
1952-53	46	29	208	527	1223	1432	1085	1070	980	756	608	240	8204

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1953-54	0	0	177	589	900	1469	1273	981	1067	621	352	284	7733
1954-55	20	96	311	730	890	1492	1748	1464	1272	809	462	201	9495
1955-56	50	8	310	646	1168	1256	1309	1514	1111	619	302	145	8438
1956	21	83	235	673	1178	1373							

MONTHLY AND SEASONAL SNOWFALL

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1949-50	-	-	-	-	-	-	9.6	6.5	7.5	0.8	T	.0	-
1950-51	.0	.0	0 1	T	0.2	5.0	7.6	5.6	3.3	T	0.2	.0	22.0
1951-52	.0	.0	.0	T	3.0	7.2	6.1	10.0	4.1	T	T	.0	30.4
1952-53	.0	.0	.0	.0	0.9	6.8	1.3	0.1	7.0	T	T	.0	16.1
1953-54	.0	.0	.0	T	T	3.1	1.1	3.2	3.9	T	.0	T	11.3

Season	July	Aug.	Sept.	Oct.	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May	June	Total
1954-55	.0	.0	T	T	.0	2.6	8.1	3.6	1.3	3.8	4.7	.0	24.1
1955-56	.0	.0	.0	0.1	1.6	6.0	10.5	4.8	4.1	T	T	.0	27.1
1956	.0	.0	.0	2.4	3.1	1.5							

STATION LOCATION

Location	Occupied from	Occupied to	Airline distance and direction from previous location	North Latitude	West Longitude	Elevation above Sea level: Ground	Elevation above Sea level: Actual barometer elevation (ft.)	Elevation above Ground: Wind instruments	Elevation above Ground: Extreme thermometers	Elevation above Ground: Psychrometer	Elevation above Ground: Telepsychrometer	Elevation above Ground: Tipping bucket rain gage	Elevation above Ground: Weighing rain gage	Elevation above Ground: 8" rain gage	Elevation above Ground: Hygrothermograph	Elevation above Ground: Dew Cell	
IDAHO FALLS 46 W	1/ 1/50	PRESENT		43° 32'	112° 57'	4933	4939	20 250a	6	5	5b		5	4	5	5	(a)On Radio tower. (b) ">. stalled 6/22/51.

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Unless otherwise indicated, dimensional units used in this bulletin are: temperature in degrees F.; precipitation and snowfall in inches; wind movement in miles per hour; and relative humidity in percent.

Record mean values at the end of the Average Temperature and Total Precipitation tables are long-term means based on the period of record beginning in 1950. Values have not been corrected for changes in instrument location listed in the Station Location table.

Sky cover is expressed in a range of 0 for no clouds or obscuring phenomena to 10 for complete sky cover. The number of clear days is based on average cloudiness 0-3 tenths; partly cloudy days on 4-7 tenths and cloudy days on 8-10 tenths. Monthly degree day totals are the sums of the negative departures of average daily temperatures from 65°F. Sleet was included in snowfall totals beginning with July 1948.

Heavy fog in the Means and Extremes table also includes data referred to at various times in the past as "Dense" or "Thick".

The upper visibility limit for heavy fog is 1/4 mile.

Below zero temperatures are preceded by a minus sign.

* Less than one half.
\- No record.
\# Also on earlier dates, months, or years.
T Trace, an amount too small to measure.

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NARRATIVE CLIMATOLOGICAL SUMMARY (Continued)

during the period of record. Conditions have varied from winters having practically no snow at all to the opposite extreme (winter of 1951-52) in which snow reached a depth of one to two feet over all the NRTS, causing near paralysis of construction activities and transportation. Large expenditures were made for snow removal, which was an almost daily task. Blowing or drifting snow is by far the greatest hazard and general nuisance of any weather phenomenon that occurs in the region. Most blowing snow is driven by south-southwesterly winds.

Destructive winds have not been a problem. Tornadoes have never been reported in the region, although funnel-shaped clouds have been sighted over the NRTS on several occasions. Structures designed to withstand average winds of 60 m.p.h. and peak gusts to 80 - 85 m.p.h. are not likely to be damaged.

Four to six thunderstorms can be expected during each summer month and some of these produce hail that can be damaging to crops. Lightning protection is a must for all man-made structures, since the lack of comparable natural targets makes them particularly vulnerable.

Super-cooled fog or low stratus clouds occur occasionally in winter with a snow cover and a persistent high pressure system, and may last for several consecutive days. These conditions permit the accumulation of considerable rime icing on power and telephone lines as well as on trees. Destructive glaze icing has not occurred in this area during the period of record.

The dust content of the air is very small while the ground is snow covered. Dust-favoring winds (gusts of 20 m.p.h. or more) occur on the average for 6 hours during 13 windy days per month. Dust concentrations have varied from a low of 0.014 mg/m^3 over snow cover to a high of 0.77 mg/m^3 on a day when dust devils were present near the sampling station. Concentrations vary widely from place to place, depending on human activity. Average daytime concentrations during the warmer months run about 0.07 mg/m^3. Visibility is not significantly reduced by dust except briefly and infrequently during dry thunderstorms.

Weathering of concrete and hard surface roads and streets is a very serious problem, mainly due to the large number of freeze-thaw cycles during all but summer months. Precautions must also be taken against adverse effects of excessive cold in winter and extreme dryness in summer when pouring concrete.

NWRC, Asheville, N.C. --- 2/2/57 --- 750

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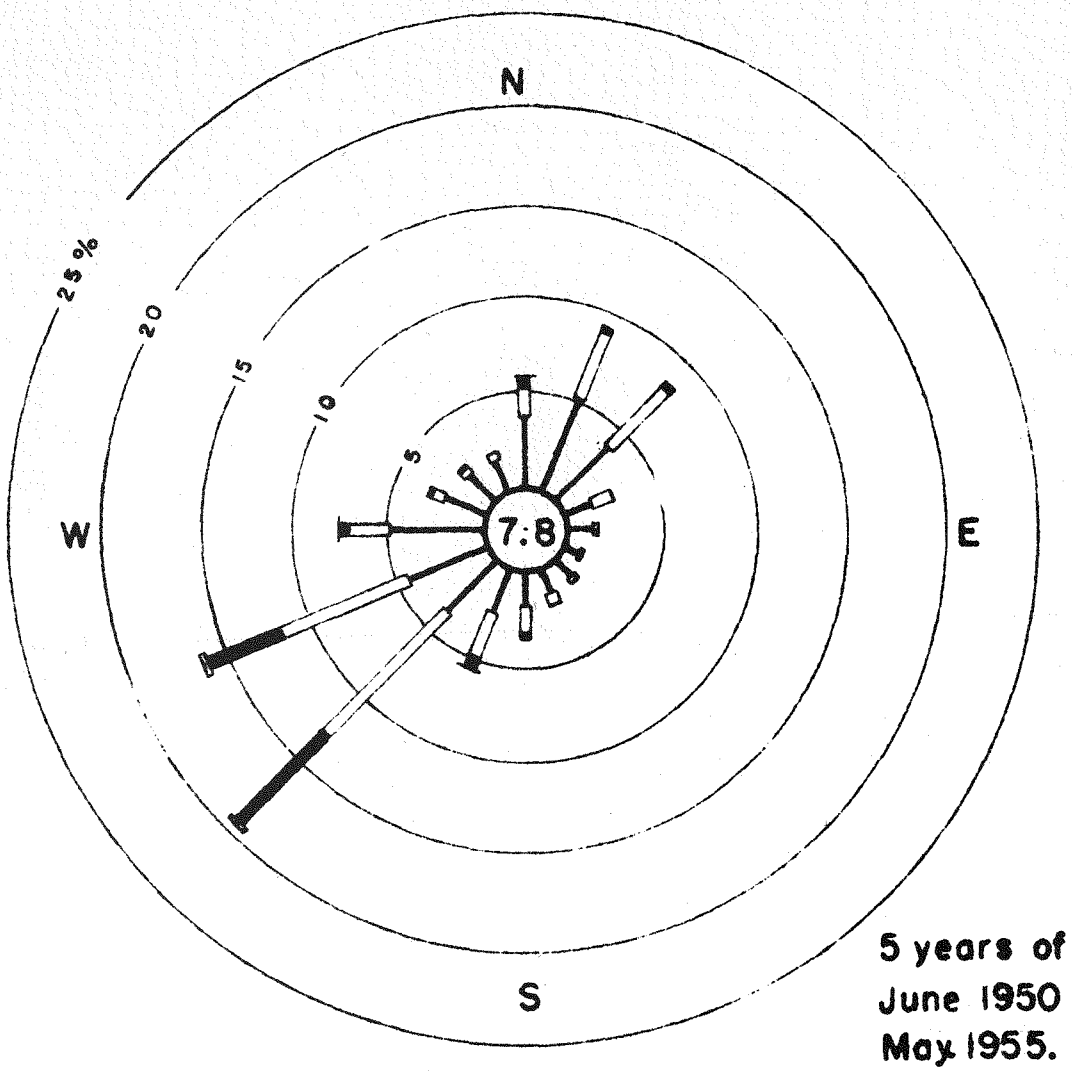
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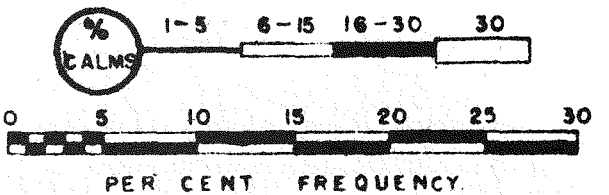
ANNUAL WIND ROSE

Central Facilities (WBO), N.R.T.S., IDAHO — 20-Ft. Level Wind



5 years of record
June 1950 thru
May 1955.
370 hours missing.

MILES PER HOUR



U.S.W.B.

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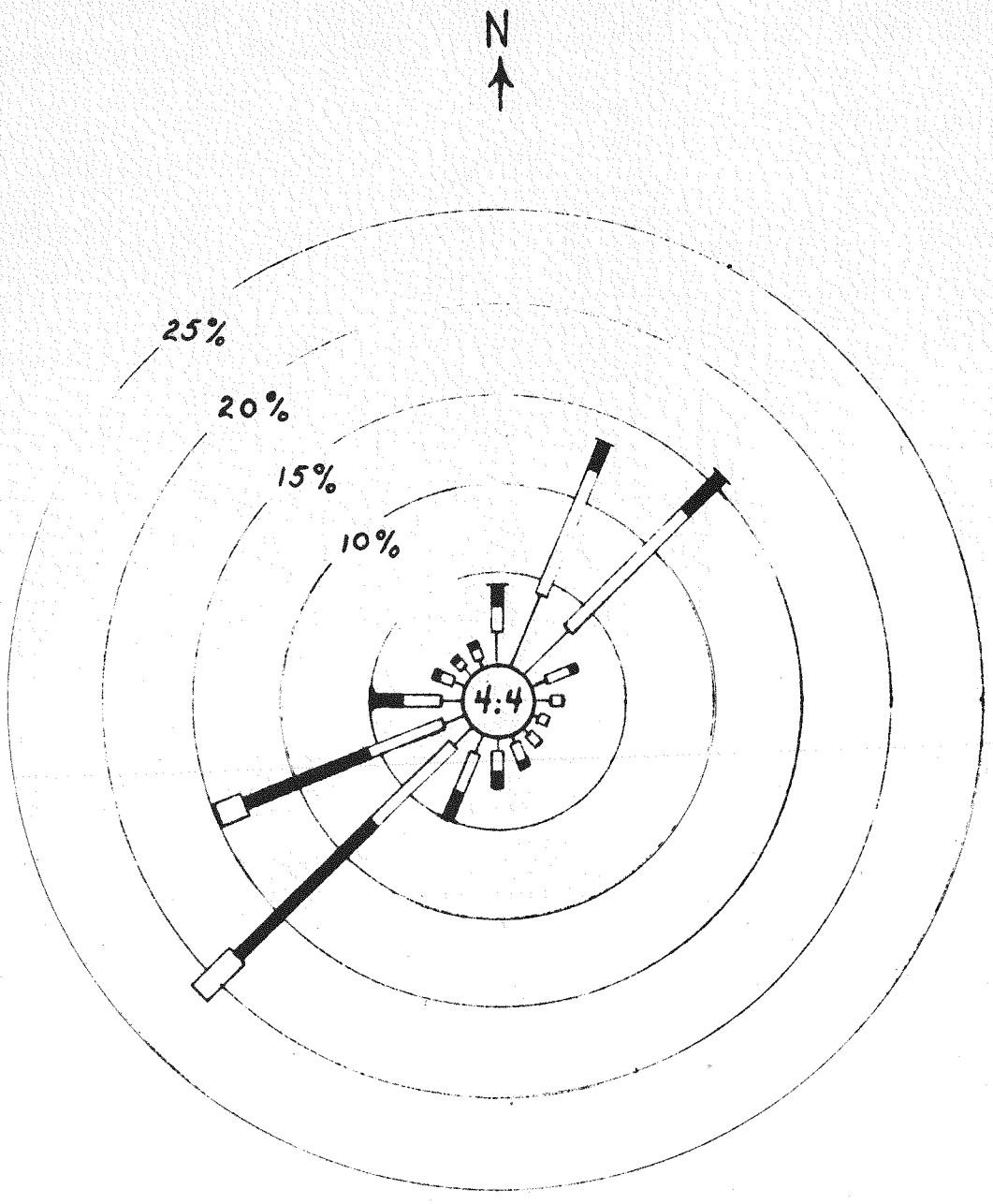
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WIND ROSE
250-FOOT LEVEL
CENTRAL FACILITIES AREA
NRTS, IDAHO

JULY 1951 THRU JUNE 1955



34,516 HOURS OF RECORD
548 HOURS MISSING

MILES PER HOUR
% CALMS
1-5
6-15
16-30
>30

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U.S.W.B.
Idaho Falls, Idaho
2/56

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APPENDIX E

AEROJET NUCLEAR TECHNICAL STANDARDS

This appendix shows the Index to Aerojet Nuclear Technical Standards (ANTS) and, as indicated by an asterisk (*), actual standards selected for applicability to the Hazard Summary. The selected standards are typical in format and coverage to those that are not included.

TECHNICAL STANDARD

GAS COOLED REACTOR EXPERIMENT-I

INDEX

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*102	Reactor Operation
*103	Demineralized Water Loop Operation
*104	Cooling Water Loop Operation
*105	Safety Circuit System Operation
*106	Annunciator System Operation
*107	Reactor Flooding Operation
*108	Pool Water System Operation
109	Modification to Class BB, D & E Piping

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TECHNICAL STANDARD

GAS-COOLED REACTOR EXPERIMENT - I

SYSTEM OR EQUIPMENT

Primary coolant loop and associated equipment.

STANDARDS

1. Flow

a. The system shall be operated in such fashion that the flow as measured by the mass flow meter (FE-101) does not exceed 70,000 lbs per hour.

b. A circulation rate of at least 6,000 lbs per hour shall be maintained at all times when the reactor power level, as indicated by any power level indicator, is greater than 500 watts.

2. Temperature

a. The temperature of the circulating coolant stream shall never be permitted to rise above 1250°F at any sensing location in the system.

3. Composition

a. The gas circulating in the primary coolant loop shall have the following composition before the reactor power is raised above 500 watts as indicated on any power level indicator.

	Min.	Max.
Nitrogen	99.5% Shutdown	---
Oxygen	0.1%	0.6% (Shutdown)
Humidity		0°F Dewpoint (Shutdown)

4. Pressure

a. The pressure in the circulating coolant stream shall never exceed 220 psig as measured at any sensing location in the system.

5. Activity

a. The primary coolant system shall not be operated when the radio-activity level as measured by the fission product monitor or any of the loop monitors:

1. Increases by a factor of 5 or more in 15 minutes or less.

2. Exceeds 50 r/hr.

6. E-101 Primary Coolant Heat Exchanger

a. This equipment shall be operated to conform to the following standards:

(1) Exit gas temperature shall not exceed 150°F. See TS-103 concerning water side of this exchanger.

7. C-101 Primary Coolant Compressor

a. This equipment shall be operated to conform to the following standards:

(1) Seal water reservoir shall contain at least 70-gallons of demineralized water.

(2) Seal water pressure shall be at least 250 psig.

(3) The back pressure on the seal system shall be maintained by adjusting FIC-103 to pass not more than 8 lbs/hour.

(4) Lube oil temperature exiting from the cooler shall not exceed 120°F.

(5) Cooling water temperature from the lube oil cooler shall not exceed 85°F.

(6) Lube oil system pressure shall not exceed 300 psig.

(7) Cooling water system pressure for lube oil cooler shall not exceed 75 psig.

(8) The pressure drop across the lube oil filter shall not exceed 10 psig.

(9) The lube oil reservoir shall not contain less than 160 gallons of oil.

(10) The gas temperature at the suction shall not exceed 150°F.

8. FH-101 Fired Heater

a. This equipment shall be operated to conform to the following standards:

(1) Tube wall temperatures shall not exceed 1275°F.

(2) The atomizing air pressure shall be at least 100 psig.

(3) The fuel oil pressure shall be at least 115 psig.

9. F-101-A&B Gas Filters

a. This equipment shall be operated to conform to the following standards:

(1) The flow rate as controlled by FIC-105 shall not exceed 135 SCFM.

(2) The gas temperature shall not exceed 200°F.

(3) The filter shall be changed when the flow as indicated by FIC-105 is reduced to 55 SCFM or when the activity, as shown by RM-101 reaches 10 R/hour, whichever occurs first.

10. C-102 Shutdown Cooling Blower

a. This equipment shall be operated to conform to the following standards:

(1) The pressure, as indicated by PR-102, in the reactor exit duct shall not exceed 15 psig.

(2) The inlet temperature as indicated by TI-111 shall not exceed 300°F.

11. E-103 Shutdown Cooler

a. This equipment shall be operated to conform to the following standard:

(1) The inlet gas temperature, as measured by TE-103 shall not exceed 1250°F.

(2) The exit gas temperature, as measured by TI-111, shall not exceed 300°F.

(3) Pressure in the system as measured by PR-102 shall not exceed 15 psigs.

TECHNICAL STANDARD

GAS-COOLED REACTOR EXPERIMENT-I

SYSTEM OR EQUIPMENT

Reactor

STANDARDS

1. Reactor Startup

a. The reactor shall not be made to approach a critical state unless all of the following conditions are satisfied:

(1) All control rods shall be operable.

(2) A neutron source shall be in place so that a significant counting rate is observable on the start-up counting instrumentation.

(3) All neutron detection instrumentation shall be operable (Log N-period systems, composite safety amplifiers, etc.)

(4) Moderator shall be in place, at least to the level of the top tube sheet.

(5) The initial operation in the approach to the critical state shall be the withdrawal and "cocking" of the safety rods. This standard shall apply whether criticality is to be attained by the addition of fuel to the core, by the withdrawal of rods from a known super-critical core, or by flooding.

(6) All safety circuits shall have been demonstrated to be operative within four hours of the initiation of the start-up.

(7) Reactivity shall not be added to the reactor in a subcritical state in increments which represent more than one half of the difference between the existing k_{eff} and the predicted critical k_{eff} (equals 1.0). This restriction does not apply above k_{eff} of 0.995.

(8) Reactivity shall not be added to the reactor in a subcritical state at a rate which exceeds 1.5% Δ K/K/minute. This restriction does not apply above k_{eff} of 0.995.

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(9) Once a k_{eff} of 0.995 has been attained, reactivity shall not be added at a rate which exceeds 0.1% Δ K/K/minute. Controls shall be established to permit definitive identification of the 0.995 condition.

2. Reactor Operation

a. The power level of the reactor as calculated from the mass flow rate and over-all temperature rise (FE-101 & TdI-102) shall not be permitted to rise above 2300 kilowatts.

b. Operation shall be discontinued if any of the following conditions occur:

(1) A scram signal occurs but the automatic scram system does not operate. In this case a manual scram shall be initiated.

(2) Either safety rod or more than one of any of the rest of the rods is known to be inoperable.

(3) Neutron detection channels III and IV (Log N-Period systems) are known to be inoperable. This standard applies only to the control components of the system and exrludes the period recorder and the Log N Recorder.

(4) Any indicator or recorder indicates a condition which should have initiated an automatic scram. In this case a manual scram shall be initiated.

(5) Either of the low count rate systems fail provided nominal operating power level is less than 100 watts.

(6) More than one of the four safety channels (V, VI, VII or VIII) is known to be inoperable. This standard applies only to the control components of the system and does not include the recorder.

c. Reactivity shall not be added to the reactor when critical or super critical at a rate which results in a period shorter than 10 seconds.

d. The reactor shall not be operated when the critical configuration is such that more than twelve inches of the fine rod is inserted with the shim rods fully inserted and all other rods out.

3. Reactor Shutdown

a. Normal shutdown of the reactor shall be accomplished in such a manner that the rate of change of outlet primary coolant temperature does not exceed 50 degrees Fahrenheit per minute.

TECHNICAL STANDARD

GAS-COOLED REACTOR EXPERIMENT - I

SYSTEM OR EQUIPMENT

Demineralized Water Loop and associated equipment.

STANDARDS

1. Flow

a. The system shall be operated in such a fashion that the flow, as indicated on FI-104, does not exceed 1000 gpm.

2. Temperature

a. The temperature of the circulating water shall not be permitted to rise above 205°F as measured on TI-108.

3. Conductivity

a. The conductivity of the circulating water shall not be permitted to rise above 10 micromho as measured in a sample drawn from the system.

4. Pressure

a. The pressure in the system shall not be permitted to rise above 170 psig as measured at any pressure sensing element in the system.

5. Makeup - Conductivity

a. The conductivity of water used as make-up to the system shall never exceed one micromho.

6. NPSH - Circulating Pumps

a. Adequate NPSH for the circulating pumps shall be assured by pressurizing the demineralized water surge drum to at least 30 psig.

TECHNICAL STANDARD

GAS-COOLED REACTOR EXPERIMENT - I

SYSTEM OR EQUIPMENT

Cooling water loop and associated equipment.

STANDARDS

1. Temperature

a. The system shall be operated in such a fashion that the temperature of the cooling water at the exit from the water to water exchanger does not exceed 100°F.

b. The system shall be operated in such a fashion as to prevent any icing on the cooling tower.

c. The system shall be operated in such a fashion that the minimum temperature in the cooling tower basin is 40°F.

d. The system shall be operated in such a fashion that the pump outlet temperature as indicated by TI-302 does not exceed 75°F.

2. Pressure

a. The pressure in the system as measured at any sensing location shall not exceed 75 psig.

3. Treatment

a. Adequate additions shall be made to the water in the system to maintain the following limits:

pH - 6.5 - 8.0

Residual Chlorine - max-10ppm; min-5 ppm

TECHNICAL STANDARD

GAS-COOLED REACTOR EXPERIMENT-I

SYSTEM OR EQUIPMENT

Reactor Safety Circuits

STANDARDS

1. Safety circuits will be checked before each reactor start-up except recovery from a scram where there was no indication of malfunction of the safety circuits.

2. Safety circuits may be disabled only for the following reasons and periods with the indicated authorization.

a. Manual scram - may never be disabled.

b. Power failure scram - may be disabled for up to two hours. For relay checking only with the approval of the Operating Superintendent.

c. Gas flow low scram - may never be disabled.

d. Gas outlet temperature high scram - may never be disabled.

e. Power level high scram - may never be disabled.

f. Seismometer scram - may be disabled for up to two hours for adjustment only with the approval of the Operating Superintendent.

g. Reactor room doors scram - may be disabled for up to 15 minutes to permit entry and exit only during operation with the approval of the Shift Supervisor.

h. Run-safe switch scram - may be disabled for up to four hours for maintenance with the approval of the Shift Supervisor. NOTE: Reactor Room Doors may not be disabled while Run-Safe is disabled and vice versa.

i. Log-N high scram - may be disabled for up to two hours for maintenance with the approval of the Shift Supervisor.

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j. Period short, channels I, II, III, IV- scram may be disabled for up to two hours for maintenance with the approval of the Shift Supervisor.

k. Period short, channels III and IV scram - may never be disabled.

l. Safety rods pressure, low scram - may never be disabled.

m. Manual setback - may never be disabled.

n. Period short, channels I and II setback- may be disabled at operating levels above 150 watts with the approval of the Shift Supervisor.

o. Gas outlet temperature high setback - may be disabled for maintenance with the approval of the Shift Supervisor.

p. Gas flow low setback - may be disabled for maintenance with the approval of the Shift Supervisor.

q. Power level high setback - may be disabled for maintenance with the approval of the Shift Supervisor.

TECHNICAL STANDARD

GAS-COOLED REACTOR EXPERIMENT-I

SYSTEM OR EQUIPMENT

Annunciator System

STANDARDS

a. All annunciator lights shall be tested at least once per shift and repairs made immediately.

b. Annunciator circuits are divided into three categories for purposes of definition under operating rules (Classes A, B & C).

1. Reactor operation may not be initiated or continued with any Class A annunciator known to be inoperative except where the trouble is known to be a burned out light bulb. In this case, the bulb must be replaced within 15 minutes of the time of discovery.

2. Reactor operation may not be initiated with any Class B annunciator known to be inoperative. Operation may continue for up to four hours from the time of discovery provided active work is in progress to correct the deficiency. During this period surveillance of the backup instrument is required.

c. Annunciators are classified as follows:

CLASS A

Panel	Description
A-1, A-2, A-3	Heater Burner - Flame Failure
B-1	Emergency Power Supply - Failure
B-4	Blow-down N_2 - Low Pressure
B-6	Main Compressor or Gear - Exit Oil - High Temperature
B-8	Fuel Element - High Temperature
B-5	Compressor - Low Speed
B-9	Main Loop Makeup - High flow
C-1 thru C-16	All scram and setback annunciators
D-12	Demineralized Water Surge Drum - Low Level
D-13	Scram annunciator
D-6	Demineralized Water System - High Temperature
D-10	Reactor Pool - Low Level
E-1	Gas Heat Exchanger - Outlet Gas Temperature High
E-2	RAMS - High activity

Panel	Description
E-3	Nuclear Instrument Failure
E-4	Water in Main Loop
E-5	T/C Manifold - High Pressure
E-6	Inlet Duct - High Temperature
E-7	Outlet Duct - High Temperature
E-8	FPM High
E-9	Water in Rods
CLASS B	
A-4	Fuel Oil Supply - Low Pressure
A-6	Utility Water System - Low Pressure
A-7	Utility Air System - Low Pressure
A-11	Steam Boiler - Flame Failure
A-13	Pool Water Deionizer - High Conductivity
A-14	Raw Water Deionizer - High Conductivity
A-15	Cooling Tower Basin - Low Level
B-7	Main Compressor Lube Oil System - Low Pressure
B-14	Main Compressor Seal Water System - Low Pressure
B-15	Main Compressor - Oil or Water Reservoir - Low Level
B-10	Main Loop - High Humidity
B-11	N_2 Storage - Low Pressure
B-16	Main Loop - O_2 Content High
B-12	O_2 Storage - Low Pressure
B-13	Instrument Air - Low Pressure
D-1	Main Loop - Low Flow
D-2	Main Loop - High Pressure
D-3	Main Loop - Low Pressure
D-4	Reactor Outlet Gas Temperature - High
D-5	Demineralized Water System - Low Flow
D-7	Water Heat Exchanger - Cooling Water Outlet Temperature - High
D-8	Cooling Water System - Low Pressure
D-16	Demineralized Water Surge Drum - Low Pressure
CLASS C	
A-8	Fuel Oil Storage Tank - Low Level
A-5	Utility Water Storage Tank - Low Level
A-9	Utility Water Storage Tank - High Level
A-12	Boiler Fuel Tank - High Level
A-16	Cooling Tower Basin - High Level
B-2	Shutdown Cooling Loop - Low Flow
B-3	Shutdown Cooling Loop - High Temperature
D-9	Waste N_2 Storage Tank - High Level
D-11	Reactor Pool - High Level
D-14	No.3 Waste Tank - High Level
D-15	No.2 Waste Tank - High Level

TECHNICAL STANDARD

GAS-COOLED REACTOR EXPERIMENT - I

SYSTEM OR EQUIPMENT

Reactor flooding system.

STANDARDS

1. Reactor flooding shall never be undertaken unless a low level counting system is in operation and under surveillance.

2. Reactor flooding shall never be undertaken unless a source is present in the reactor to yield a significant counting rate.

3. Reactor flooding shall never be undertaken without certain knowledge that the dump valve is operative.

4. Flooding of the bottom plenum may be undertaken at any reasonable filling rate provided the system has been calibrated to guarantee that flooding will automatically cease when the water level is four and one-half inches below the bottom tube sheet.

5. Flooding of the pressure tube portion of the reactor may be performed only through the restrictive orifice at a flow-rate not to exceed two gallons per minute.

6. Flooding of the upper plenum may be performed at any reasonable rate provided the low level counting rate system has shown that additional flooding will not alter the neutronic characteristics of the system.

7. Flooding shall be accomplished with demineralized water which may be heated if necessary.

8. Flooding shall not be undertaken when the differential temperature between flooding water and any fuel element temperature is greater than 250°F.

TECHNICAL STANDARD

GAS-COOLED REACTOR EXPERIMENT - I

SYSTEM OR EQUIPMENT

Pool water circulating system and associated equipment

STANDARDS

1. Pressure

a. The system pressure shall not exceed 75 psig as indicated by any sensing element in the system.

2. Temperature

a. The system shall be operated in such a fashion that the temperature of the water in pool as indicated on TIC-201 does not exceed 105°F.

3. Conductivity

a. The conductivity of the circulating system shall be maintained below one micromho.

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APPENDIX F

AEROJET NUCLEAR STANDARD OPERATING PROCEDURES

This appendix shows the Index to Aerojet Nuclear Standard Operating Procedures (ANSOP) and, as indicated by an asterisk (*), actual procedures selected for applicability to the Hazards Summary. The selected procedures are typical in format and coverage to those that are not included.

AEROJET NUCLEAR STANDARD OPERATING PROCEDURES
GAS-COOLED REACTOR EXPERIMENT-I

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GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 2 - RADIOLOGICAL SAFETY

TITLE: RADIOLOGICAL SAFETY - GENERAL

SCOPE

This procedure outlines the general considerations and ground rules which represent the basis of the radiological safety program at the GCRE-I.

PROCEDURE

1. Responsibility

The basic responsibility for the safety of each individual rests with the individual. Systems are established and controls set up to assist him to discharge this responsibility but no system or control is adequate without the thinking cooperation of the individual and a sincere interest in his own welfare.

The line organization has the prime responsibility for adherence to procedures and controls. They have the authority to administer disciplinary action where indicated to enforce the procedures and controls.

The Health & Safety Officer has the responsibility for continuous cognizance and evaluation of the program. His observations and recommendations will be acted on by the line organization to improve performance. His clearance of sensitive procedures and policies is required.

2. Philosophy

The basic philosophy of the radiological safety program is that no individual shall receive any exposure that is not required and justified in the execution of his responsibilities. Where exposure is required, the conditions under which the dosages is received will be controlled so that there is exact knowledge of the exposure. The controls established in the program are within the letter and intent of the regulations established in the AEC Manual #1, Chapter Series 0500.

3. Incident Investigations

An incident is defined as either of the following:

a. The discovery of an unanticipated radiological hazard or the discovery of an unanticipated increase in degree of a recognized radiological hazard.

b. The recognition of a deficiency in establishment of or adherence to the controls which permitted a contamination spread or over-exposure or which resulted in a condition of potential contamination spread or overexposure.

An incident will be formally investigated by a committee appointed by the Assistant Project Manager, chairmanned by him and including the Health & Safety Officer. The investigation will determine the cause of the incident and establish the action to be taken to prevent its recurrence. The result of the investigation will be published and the Assistant Project Manager has the responsibility for implementing the action to prevent recurrence.

4. Training

The line organization with the assistance of the Health & Safety Officer shall institute and implement a training program in radiological safety. This program shall be comprehensive enough to impart to the trainees a fundamental knowledge of the basic hazards and a detailed knowledge of the procedures and controls which are associated with the program.

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GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 2 - RADIOLOGICAL SAFETY

TITLE: RADIOLOGICAL HAZARDS CONTROL

SCOPE

This procedure sets forth the basic control criteria which will be used to minimize the radiological hazard exposure of personnel associated with the GCRE-I. The radiation exposure and contamination limits are generally more conservative than those specified in AEC Manual 1, Chapter 0524. Adequate records and controls have been established (see ANSOP 2100 and 2900) to assure that the procedures herein described have been satisfied.

A. MAXIMUM PERMISSIBLE EXPOSURES TO EXTERNAL SOURCES OF RADIATION

1. Total Exposure - No more than 50 mrem/day or 300 mrem/week or 15 Rem/year, provided that the accumulated lifetime exposure of an employee with age = N does not exceed 5 (N-18) REM.

Special Notes:

a) Up to 900 mrem may be accumulated in one week with prior approval of the Health & Safety Officer and Operating Superintendent. This may occur only once in any calendar quarterly period.

b) The entire weekly dose may be accumulated in one day with prior approval of the Health & Safety Officer and Operating Superintendent.

c) Up to 100 mrem may be accumulated in one day with prior approval of the Shift Supervisor. This may occur only once in any week. Notice must be given to the Health & Safety Officer of such action within 24 hours.

d) In all of above cases the Shift Supervisor must contact the individuals involved and go over with them their exposure history, being sure they understand how to control their activities for the balance of the week or quarter.

e) In no case will an individual be permitted to accumulate more than 300 mrem in a single exposure.

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f) The film badge record will be the prime source of information pertaining to radiation exposures.

2. The table given below may be used to correlate indicated radiation levels with the MPE established under #1.

Radiation	Rate to give 50 mrem in 8 hrs	Rate to give 300 mrem in 40 hrs
X, or B	6.25 mr/hr	7.5 mr/hr
slow neutrons	1250 $n/cm^2/sec$	1500 $n/cm^2/sec$
fast neutrons	42 $n/cm^2/sec$	50 $n/cm^2/sec$

B. MAXIMUM PERMISSIBLE CONCENTRATION OF RADIOACTIVITY IN AIR AND WATER

1. In accordance with AEC Manual Chapter 0524-02-2 C, the concentration of airborne or water borne radioactive contaminants (where inhalation or ingestion is feasible) will be kept within the limits established in NBS Handbook #52. The added instruction that internal emitters having the whole body or the gonads as the critical organ will be kept to one-third the MPC listed in NBS #52 will also be met. For unidentified mixtures of contaminants in air or water, the following MPC's will apply.

Medium	$\beta\gamma$ µc/ml	α µc/ml
AIR -----	10^{-9}	5×10^{-12}
WATER ---	10^{-7}	10^{-7}

2. Frequent checks will be made of the GCRE-I environment to establish the airborne and water borne radio-contaminant concentration level. The presence of contamination levels greater than those listed in (1) above will require corrective and personnel protection measures until such time as allowable levels have been reestablished.

C. SURFACE CONTAMINATION

1. Any area known to be contaminated will be isolated. Following isolation the hazard will be evaluated and decontamination proceedures will be initiated.

2. One of the following decontamination procedures will apply to every contaminated item or surface discovered at the GCRE-I:

a. Portable items that cannot be profitably decontaminated will be packaged in a manner suitable for disposal and removed from the site.

b. Immovable items or articles of high intrinsic value will be decontaminated by chemical means to a point that no loose surface contamination remains and (where possible) the level of residual "fixed" contamination is less than or equal to the radiation background in the area of the item.

c. Wherein it may be impossible to reduce the residual "fixed" contamination level below a significant point, the area of contamination will be painted or otherwise protected from abrasion, it will be posted or marked with an "under paint contamination" (UPC) sign and the surface dose rate will be indicated. Area control pertinent to this procedure is covered in ANSOP 2200.

D. PERSONNEL RADIOLOGICAL HYGIENE

1. As part of the employee pre-employment physical examination, a complete blood analysis and vital capacity measurement will be taken.

2. After employment, the "lab work" and vital capacity measurement will be repeated no less frequently than every nine (9) months.

3. A follow-up physical examination will be conducted no less frequently than each eighteen (18) months after employment.

4. In the event of a serious incident, or where otherwise indicated, bio-assays will be made of biological specimens supplied by GCRE employees to determine the presence or absence of radioactive material.

5. All data accumulated under items 1 - 4 above will be reviewed and evaluated by Aerojet's expert in Radiological Medicine, who will advise GCRE management of corrective actions when necessary.

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PROCEDURE SECTION: 5 Reactor Operation

TITLE: Special Work Permit

SCOPE

This procedure outlines the operation of the Special Work Permit System. This system has been established as the basic control of the radiological hazard protection program.

In general, the Special Work Permit (SWP) establishes the controls under which a person may enter a radiologically hazardous area (generally called a Radiation Danger Zone). It is prepared in advance and signifies that an evaluation of the hazard has been made by some person other than the one performing the work and that the evaluation and restrictions are understood by the person performing the work.

For the purposes of this procedure and the radiological hazards protection program in general, the following definitions are established:

1. Special work Permit (SWP) ANSOL 2350. The form which, when properly executed authorizes entrance to a radiologically hazardous area.

2. Radiation Danger Zone (RDZ) - an area or piece of equipment which represents an existing or potential radiological hazard. An RDZ is defined by barriers, either permanent (handrail, equipment surfaces, etc.) or temporary (portable standards with yellow and magenta rope) or by easily identifiable markings on the floor (yellow and megenta tape or paint). The boundaries of an RDZ are also marked at appropriate intervals by yellow and magenta octagonal signs reading "Danger - Radiation Zone". See ANSOL 2300 for rules concerning establishment of RDZ.

3. Maximum Permissible Exposure - that total amount of radiation, corrected to standard units, which an individual is permitted to receive in a specified period of time. See ANSOP 2100 for applicable maximum permissible exposures and other information on this subject.

4. Dosage Rate - the rate, expressed in mrem/hr (See ANSOP 2100) to which an individual is exposed from a given source at a given place. In SWP work, more specifically, the maximum rate of exposure for performance of the specific job described on the SWP.

5. Time Limit- the maximum time which the individual may remain in the radiation field without receiving more than the maximum permissible exposure.

6. Protective Equipment - the clothing and other protective equipment which the SWP requires the individual to use in the performance of the job, either for his protection or for the prevention of the spread of contamination.

7. Survey Instruments - The devices available for the detection and measurement of dosage rates or exposures. Specifically, in the SWP system, the portable measuring instruments which an individual may be required to use on the job or the personal exposure monitoring equipment, such as film badges and dosimeters, which he may be required to carry and read.

8. Radiation Exposure Record (ANSOL 2351). The card for each individual on which he keeps a cumulative record of his exposure on each SWP for the week. This record is to aid in preventing his exceeding the maximum permissible weekly exposure.

PROCEDURE

A. PROCEDURE FOR COMPLETING THE SWP

1. The individual who will perform the work in the RDZ is responsible for initiating the SWP. Exceptions to this rule are: (a) SWP may be prepared in advance for scheduled work, in which case the person performing the work will find the SWP completed except for authorizing signature. (b) SWP for visitors will always be prepared by a member of the regular operating crew.

2. The person preparing the SWP will complete the following sections:

a) The file number - SWP will be numbered in series for filing, identification, and control. Consult the SWP log (ANSOL) and log in your SWP. Assign to it the next number in sequence.

b) The date - fill in the date on which the SWP will be used initially.

c) Job description - insert information concerning the specific nature of the work to be performed and the area within which the work will occur. Be sure that enough information is given to permit complete evaluation of the hazard.

d) Survey data - fill in the exposure rate and time limit if these data are known. If the data are not available, check the block under Survey Requirements which indicates that these data are to be determined on the job before the start of work.

e) Survey requirements - if the job is such that dosage rates are high or if the rates may change during the course of work because lines are opened, shielding is removed, etc., check continuous monitoring. If it appears that continuous monitoring is not justified but that occasional checks of dosage rate should be made, check intermittent monitoring and specify in the space provided when such monitoring is to be accomplished. If any aspect of the job requires special monitoring, so indicate in the space provided.

f) Protective clothing - check the appropriate blocks to provide adequate protection against personal contamination.

3. Present the SWP to the Shift Supervisor (or in his absence the Operating Superintendent) for approval. The authorizing supervisor will read the permit to satisfy himself that the hazard has been properly evaluated before he signs the SWP. In addition if the time limit is known to be less than 30 minutes or expected to be less than one hour, the Health & Safety Officer must also authorize the SWP.

4. SWP may be written to authorize work on a continuing job for up to 24 hours. If the SWP is to be used on shift following the one on which it was originally authorized, it must be reviewed and re-authorized by the Shift Supervisor of the following shift before work can begin on that shift. Particular attention shall be given to exposure rate data since this may change in the course of time.

B. PROCEDURE FOR USE OF SWP

1. Place the authorized SWP on an SWP clipboard.

2. Read the SWP and sign in "used by" blank.

3. Observe the survey requirements and, if required, proceed to the Instrument Shop and obtain a survey meter and special dosimeters as indicated. Zero and source check the survey instrument and charge the pencils according to the procedures posted.

4. Proceed to the change area and put on the protective clothing specified.

5. Sign in on the SWP and hang the SWP and clipboard on the board provided.

6. Enter the RDZ and perform the work.

7. When leaving the RDZ for any purpose, remove clothing in accordance with steps below:

* Note: Place each article in the receptacle provided.

a - Remove head covering being careful not to contaminate face or hair

b - Remove gloves without touching exterior surfaces

c - Remove breathing protective equipment

d - Remove shoe covering without touching exterior surfaces

e - Unzip coveralls and remove, touching only inside surfaces.

NOTE: If two pair of coveralls are being worn, remove the outer set before step "a" above.

f - Perform a personnel and area contamination check.

8. Sign out on the SWP and calculate exposure (if re-entry is not anticipated). If contamination is detected, call the control room on the intercom for advice and assistance with the decontamination.

9. Complete the entry on the Radiation Exposure Card.

10. If no further work is contemplated on the SWP complete the termination section, complete the SWP log, and file the SWP.

C. EXTENDED SWP

An Extended SWP may be prepared for jobs which satisfy the following requirements:

1. The job is repetitive in nature and can be adequately described and

2. The exposure rates are known to be less than 10 mr/hr or

3. The exposure rates are known to be less than 50 mr/hr and can be monitored in the course of the job.

a) Preparation of Extended SWP

1. Prepare a ditto master which will later be run onto the regular SWP form. Fill in all spaces except, in the case of 3 above, indicate "continuous monitoring" and "exposure level to be determined".

2. Submit the ditto master for authorization by the Operating Superintendent, Health & Safety Officer, and Assistant Project Manager.

3. After authorization, run copies of the Extended SWP and file in the control room.

b) Use of Extended SWP

(1) Procure a copy of the SWP and log it in the SWP log. Assign the file number to the SWP.

(2) Have the Shift Supervisor initial and date in the authorization block.

(3) Proceed as in Section B above.

STANDARD OPERATING PROCEDURE

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 3: SAFETY

TITLE: MASTER EMERGENCY PLAN - GCRE-I

SCOPE

The procedures outlined in this plan have been devised to provide the greatest possible degree of safety to GCRE-I personnel, other NRTS contractor personnel, and the public at large in the event of a catastrophe to the reactor.

The procedure is divided into the following sections:

A - General
B - Release of Radioactivity
C - Fire
D - Air Raid

PROCEDURE

A. GENERAL

1. Distribution and Cognizance

a. Each member of the GCRE-I operating organization will be required to review the Master Emergency plan no less frequently than every three (3) months in order that the operational steps will remain fresh in his mind.

b. Prior to initial criticality and no less frequently than once every six (6) months the GCRE-I Operating Superintendent will initiate an emergency drill. The emergency drill will involve a mock run-through of each of the emergency plan steps. The Shift Supervisor will make a critical evaluation of the performance of the drill and will submit his findings in writing to the Operating Superintendent.

c. All emergency services at NRTS, including the Central Facilities Fire and Ambulance Service, Central Facilities Weather Station, Central Facilities Guard Station and IDO Site Services will be supplied copies of the GCRE-I Master Emergency Plan.

d. In event of earthquake, strike or riot, the console operator will shut down the reactor and await further instructions from supervision.

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B. RADIOACTIVITY RELEASE

The emergency plan herein described will be put into effect only when all methods of controlling or averting a major release of radioactivity have failed.

1. Initiation of Alarm

a. Any member of the operating organization at the GCRE-I who feels, in his best judgment, that a major release of radioactivity has occurred, or is irrevocably imminent, is required to initiate an emergency alarm. This is accomplished by turning the Evacuation Switch on the left panel of the main console to "Long". This initiates a long warbling signal on the public address system which notifies all personnel on the site that a radioactivity release has occurred or is imminent. If the person initiating the emergency alarm is not in the control room, he shall contact the control room operator by the fastest means and direct that the alarm be sounded. The switch will be turned "Off" after two minutes.

b. Following the alarm and the completion of the emergency procedure (see below) the responsibility for direction of personnel will rest with the Shift Supervisor on duty until he is released from this responsibility by a ranking member of line supervision.

c. In lieu of any supervisor, the reactor operator who was at the control console when the alarm was initiated will be responsible for direction after the completion of the emergency plan.

2. Emergency Procedure

Upon recognition of the emergency alarm for radioactivity release personnel will take the following action:

a. The reactor operator at the control console will execute ABLE, pick up the operations log and leave the control room.

b. The second operator in the control room will record the wind direction and velocity on the ANSOL 5752, take this form and leave the control room.

c. The Shift Supervisor will obtain at least one portable survey instrument and leave the Instrument Shop.

d. Each member of the operating crew will proceed to the Counting Room by the shortest available route.

e. The first person arriving in the Counting Room will immediately turn-on the emergency radiation meter stored in that area, and will note the radiation background at the time on the data sheet provided with the instrument.

f. The ranking supervisor will attempt to call the IDO Security Dept. Dispatcher and inform them of the emergency and the radiation level in the counting room.

(1) If the call is completed, subsequent personnel rescue procedures will be as directed by the IDO Security Department.

(2) If the call is not completed, proceed with step g.

g. The ranking supervisor will take the following action:

(1) With the survey instruments available, determine the exposure level in the room.

(a) If the level is less than 50 mrem/hr, proceed with steps to dispatch the guard and one other person to the guard house to inform Central Facilities of the incident. Write out the message to avoid confusion, stating what has occurred, where the people are, what the dosage rate is and requests for any special equipment. Advise the guard and other person to return to the counting room.

(b) If the level is greater than 50 mrem/hr, proceed with the following steps.

(2) Review ANSOL 5752 and note the last listed wind direction.

(3) Issue emergency breathing apparatus and hoods (stored in the counting room) to each evacuee.

(4) Lead the evacuees through the GCRE-I site gate by the shortest available path.

(5) Withdraw from the GCRE-I site up-wind to a point where the radiation level is less than 10 mrem/hr, as indicated by the portable survey instrument.

(6) Proceed, by the shortest safe route, to the nearest habitation and call the CF Fire Department.

3. Investigation

a. As soon as practical after the emergency, the Health and Safety Officer will initiate an investigation of the incident. The written report of this investigation will discuss the cause of the incident, the results (in terms of injury or property damage) of the incident and the recommendations to prevent recurrence. If this report cannot be published within 48 hours of the time of the incident, a preliminary report will be issued and the final report published when the investigation is complete.

C. FIRE

1. General

a. Equipment - The building is equipped with an adequate number of fire extinguishers of the proper type to fight small fires occurring in the facility. In addition hose racks are provided in the building. Four fire alarm boxes are strategically located in the building which communicate directly to the NRTS Fire Alarm System.

b. Training - Personnel at the GCRE site will be trained to fight fire under the "Auxiliary Brigade" concept. Drills and simulated fires will be conducted on a regular basis to maintain the efficiency of the crew.

2. Procedure

a. Any person observing a fire within the GCRE-I area or its immediate environs will immediately proceed to the nearest fire alarm box and initiate the fire alarm.

b. The person initiating the alarm will contact the control room by the fastest possible means and request the control operator to shutdown the reactor and notify the Shift Supervisor.

c. The person initiating the alarm will then proceed, if practical, to the scene of the fire and take up his regular brigade duties.

d. Upon receipt of the information in the control room the operator will execute ABLE and announce over the public address system that a fire exists. He will specify the fire location and direct the brigade to report to the scene of the fire. He will remain in the control room unless directed by the ranking supervisor to evacuate. He will note and log the time of the alarm and contact the Assistant Project Manager and the Health and Safety Officer.

e. Upon receipt of the announcement concerning the fire, brigade members will report to the scene of the fire and fight the fire in accordance with procedures prescribed in their training program. If the fire is in a contaminated area, they will put on full protective equipment (overalls, boots, gloves, and hoods) and proceed immediately to the fire.

f. If the fire is in a Radiation Danger Zone, the control room operator will initiate the preparation of a Special Work Permit to cover the fire fighting. At the conclusion of the emergency, he will deliver the Special Work Permit to the Shift Supervisor who will have the exposure data recorded. The time of entering the Radiation Danger Zone will be assumed to be the time the alarm was sounded as noted in step d. above.

g. Upon the arrival of the C. F. fire fighting organization at the scene, the C. F. Fire Marshal will be oriented by the Shift Supervisor and will assume direction of the fire fighting at a time agreed to by he and the Shift Supervisor. The Shift Supervisor will remain with the Fire Marshal to assist as required with information peculiar to the GCRE-I facility.

h. After the emergency, the Shift Supervisor and the C. F. Fire Marshal will inspect the fire fighting equipment and have it returned to good order. The C. F. Fire Department will be notified of any extinguishers used and arrangements made for their recharging.

i. As soon as practical, the Health and Safety Officer will conduct a formal investigation of the fire. The written report of this investigation will discuss the cause of the fire, the extent of damage to government equipment, any injuries which resulted from the fire and recommend action to prevent recurrence. The Assistant Project Manager shall be responsible for the implementation of the recommendations. If the report cannot be published within 48 hours, a preliminary report will be issued and the final report published as soon thereafter as possible.

D. AIR RAID

Normally the air raid signal will be received in the control room by telephone communication from Central Facility. Upon receipt of the signal, the following procedure will be initiated:

1. Depending on the degree of urgency, the console operator will immediately execute BAKER or ABLE.

2. The other operator in the control room will turn the Evacuation Switch to "Short". This will produce a short warbling tone on the public address system which will inform all personnel at the site that an air raid is in progress. The switch will be turned to "off" after 2 minutes.

3. All personnel will proceed by the shortest path to the Control Room and remain there until the All Alear is sounded.

4. Upon receipt of the All Clear, the console operator will turn the evacuation switch to "Continuous". This will sound a continuous tone on the public address system which signals the "All Clear". The switch will be turned to "Off" after one minute.

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GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION NUMBER: 5001

TITLE: LINE SCHEDULE

SCOPE

This procedure lists all line numbers in the facility along with a short description of the function of the line. The list is divided by system for easy reference.

Information concerning material class of the various lines may be found in Specification MS-1.

LINE SCHEDULE:

GNL SYSTEM

Line No.	Function
GNL-1	Reactor to E-101
GNL-2	E-101 to C-101
GNL-3	C-101 to FH-101
GNL-4	FH-101 to Reactor
GNL-5	GNL-1 to E-103
GNL-6	E-103 to C-102
GNL-7	C-102 to E-104
GNL-8	GNL-3 to F-101 A & B
GNL-9	F-101 A & B to S-101
GNL-10	S-101 to GNL-2
GNL-11	N_2 Makeup Bank Header
GNL-12	N_2 Makeup line - small flow
GNL-13	Emer. N_2 Blowdown Header
GNL-14	E-104 to GNL-4
GNL-15	N_2 Makeup to Emer. N_2 Blowdown Cross Tie
GNL-16	N_2 Makeup line - large flow
GNL-17	F-101 A & B Drain
GNL-18	Reactor Bypass
GNL-19	Heater Bypass
GNL-21	Reactor Purge line
GNL-22	Vent line GNL-4 to GN-11 via GN-18
GNL-23	LI-102 Impulse Line
GNL-24	GNL-11 to GN-20 ("X" rod pressure)
GNL-25	Open Loop Purge Line

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GN SYSTEM

Line No.	Function
GN-1	Blowdown: GNL-2 to T-201
GN-2	C-101 Seal blowdown
GN-3	D-101 Blanket supply
GN-4	D-101 Blanket exhaust
GN-5	T-201 to C-201
GN-6	Deleted
GN-7	T-201 to stack
GN-8	Deleted
GN-9	Purge line to F. P. M.
GN-10	Vent line from F. P. M.
GN-11	Vent line from PSV 101 and 102
GN-12	Deleted
GN-13	PCV-102 Bypass
GN-14	Vent line; PSV-103
GN-15	Vent line; BD-102
GN-16	Vent line; BD-101
GN-17	Vent line; PSV-109
GN-18	Vent line from GNL-22 to GN-11
GN-19	Reactor purge vent line to GN-11
GN-20	"X" Rod pressure supply

WD SYSTEM

WD-1	E-101 to E-102
WD-2	E-102 to D-101
WD-3	D-101 to P-101 A & B
WD-4	P-101 A & B to E-101
WD-5	Loop makeup line
WD-6	S-301 Discharge
WD-7	Pool makeup line
WD-8	E-102 Bypass
WD-9	Boiler makeup line
WD-10	Deleted
WD-11	Deleted
WD-12	Reactor flooding line

WP SYSTEM

WP-1	Reactor pit overflow
WP-2	Pit to P-202 Suction
WP-3	P-202 to E-201
WP-4	E-201 to S-201
WP-5	S-201 to Pit
WP-6	Deleted

Line No.	Function
WP SYSTEM (continued)	
WP-7	Reactor drain and fill line
WP-8	WP-5 to CS-2
WP-9	SP-8 to WA-1
WP-10	E-105 Drain
WP-11	Deleted
WP-12	Reactor fill line S-104 to WP-7
WP-13	Reactor fill line WP-2 to S-104
WP-14	Reactor drain line WP-8 to sump
WP-15	Deleted
WP-16	"T" Shield cooling line to WP-2
WA SYSTEM	
WA-1	P-206 to GNL-1
WA-2	GNL-4 to P-206
WA-3	P-206 Discharge
WA-4	D-202 to P-205 Suction and drain field
WA-5	WA-10 to D-203 and WA-2
WA-6	D-203 to P-205 Suction and drain field
WA-7	S-201 to WA-3
WA-8	WA-1 to WA-9
WA-9	WA-8 to WA-5
WA-10	P-205 Discharge manifold to drain field
WA-11	Deleted
WA-12	S-201 to Resin disposal
WA-13	Deleted
WA-14	D-203 to WA-10
WA-15	D-202 to WA-10
WA-16	Deleted
WA-17	C-102 Drain to sewer
WA-18	Deleted
WA-19	Deleted
WA-20	Bypass WA-3 to WA-5
CS SYSTEM	
CS-1	D-201 to CS-3
CS-2	P-201 Discharge header to GNL-1
CS-3	P-201 Suction header from GNL-4
CS-4	E-101 Drain line
CS-5	C-101 Case drain
CS-6	FH-101 Low point drain

WC SYSTEM

Line No.	Function
WC-1	Circulating pump discharge header
WC-2	E-201 Supply
WC-3	E-201 Return
WC-4	Header to E-102 Supply
WC-5	E-102 to Return header
WC-6	Main return header
WC-7	E-301 Supply
WC-8	E-301 Return
WC-9	S-304 Supply
WC-10	S-304 Return
WC-11	C-302 Supply
WC-12	C-302 Return
WC-13	CT-301 Bypass
WC-14	E-105 Supply
WC-15	E-105 Return
WC-16	Secondary supply header
WC-17	Secondary return header
WC-18	S-303 Supply
WC-19	S-303 Return
WC-20	S-101 Supply
WC-21	S-101 Drain
WC-22	C-101 Supply
WC-23	C-101 Return
WC-24	C-301 Supply
WC-25	C-301 Return
WC-26	S-101 Return

WU SYSTEM

WU-1	P-301 to T-301
WU-2	T-301 to P-303
WU-3	WU-2 to P-302
WU-4	P-303 Discharge header
WU-5	P-302 Discharge header to WU-4
WU-6	Bldg. 610 supply line
WU-7	Return line to T-301
WU-8	Deleted
WU-9	Compressor pit flooding line
WU-10	Heater pit flooding line
WU-11	Domestic water header
WU-12	WU-4 to WP-4
WU-13	WU-12 to WA-10
WU-14	T-301 Overflow
WU-15	WU-11 to CT-301 Basin Makeup

WU SYSTEM (continued)

Line No.	Function
WU-16	CT-301 Fire Spray nozzles
WU-17	WU-12 to S-301
WU-18	S-302 to CT-301 Basin
WU-19	T-301 to PS-306 and PS-307
WU-20	WU-4 to Bldg 607 and 608 Hoses
WU-21	WU-4 to Bldg 610 Hose

AI SYSTEM

AI-1	Filter to C-302
AI-2	C-302 to E-302
AI-3	E-302 to D-302
AI-4	D-302 to S-303
AI-5	S-303 to Bldg distribution header
AI-6	S-303 Bypass
AI-7	Pressure manifold supply
AI-8	Fission chamber shield actuator supply

AU SYSTEM

AU-1	C-301 to E-301
AU-2	E-301 to D-301
AU-3	D-301 to Supply distribution header
AU-4	AU to AI Crossover
AU-5	Bldg 610 Service line
AU-6	FH-101 Supply Line
AU-7	Deleted
AU-8	S-301 Supply line
AU-9	S-301 Supply line

GO SYSTEM

GO-1	Cylinders to GNL-12

S SYSTEM

S-1	Main distribution header - boiler to separator
S-2	Separator to reactor
S-3	Reactor to condenser
S-4	S-1 to WP-9 and WP-8
S-5	WP-9 to CS-3
S-6	Supply to S-104

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LPG SYSTEM

Line No.	Function
LPG-1	D-303 to LPG-PCV-106
LPG-2	PCV-106 to FH-101
LPG-3	LPG-1 to Lab.

OF SYSTEM

OF-1	T-302 Fill line
OF-2	T-302 to P-305 A & B
OF-3	P-305-A and B Discharge header
OF-4	OF-3 to Boiler day tank
OF-5	OF-3 to Diesel day tank
OF-6	Return header to T-302
OF-7	Feed line to Mech. equipment room heater
OF-8	Return line from FH-101
OF-9	T-302 Overflow line
OF-10	Return line from Mech. equipment room heater
OF-11	Feed line to Bldg 610 heater

F SYSTEM

F-1	T-302 Blanket line

OL SYSTEM

OL-1	Main lube oil pump discharge
OL-2	Aux. lube oil pump discharge
OL-3	Lube oil cooler inlet
OL-4	Cooler to filter
OL-5	Supply header
OL-6	Return header
OL-7	Main pump suction
OL-8	OL-PSV-306 Return line
OL-9	By pass line
OL-10	OL-PCV-307 Return line

WS SYSTEM

WS-1	Main pump discharge
WS-2	Aux. pump discharge
WS-3	Cooler to filter
WS-4	Supply header
WS-5	Inner seal drain
WS-6	Outer seal drain
WS-7	Seal trap to reservoir
WS-8	Return from WS-PCV-309
WS-9	Return from WS-PSV-307

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ANSOP 5701
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STANDARD OPERATING PROCEDURE

GAS COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: SETBACK RECOVERY

SCOPE

This procedure outlines the steps to be followed in recovering from an automatic or manual setback of the reactor. The following conditions must exist before this procedure is applicalbe:

1. All neutron detection instrumentation is operative (Channels I & II may be disabled due to high count rate).

2. All elements of main loop must be in normal operation.

3. All support systems (cooling water, air, etc.) must be in normal operation.

4. All rods must be operable as evidenced by normal behavior during the shutdown and/or recent operation.

5. The elapsed time from the shutdown to the initiation of this procedure shall not have exceeded one hour.

6. No change shall have been made in the core loading or gas pressure or composition.

In principle, this procedure authorizes the rapid removal of rods to a configuration near that which prevailed at the time of the setback and then a normally cautious approach to final critical.

PROCEDURE

1. Immediately upon receipt of the automatic setback or upon initiation of a manual setback, set the rod selector on shim gang and run both sim rods full in. Also run the fine rod full in.

2. When all rods are in and power, as indicated on the neutron detector indicators, is less than 1% of pre-setback power, set the rod selector switch on setback and withdraw the setback rod. Observe the Log-N system

during this withdrawal and suspend withdrawal if the indicated level increases by morethan a factor of 1.5. If the level increases by a factor of 2 or more, trip the setback rod and repeat the operation with a slower rate of rod withdrawal.

3. Place the selector switch on shim gang and withdraw the shim rods 1/4 inch.

4. Place the rod selector switch on fine rod and withdraw the fine rod. Regulate rate of rod withdrawal so that Log-N indication does not double during this withdrawal.

5. Place the rod selector switch on shim #1 and withdraw this (followed by shim #2) at full speed until criticality is attained. Observe all nuclear instruments for evidence of a sustained period or a sustained increase in power as an indication of criticality.

6. Adjust the shim rod being used to control the period at about 30 seconds. Change the range selector switch on the linear power system to keep the recorder on scale.

7. Increase power to 150 watts and level off. Maintain this level for 5 minutes then proceed to raise power per ANSOP 5706.

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STANDARD OPERATING PROCEDURES

GAS COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: FORMAL REACTOR STARTUP

SCOPE

This procedure outlines the steps to be taken to bring the reactor to criticality under the usual conditions of reactor startup. This procedure applies to any startup where the critical configuration is known with fair precision; a startup following a maintenance outage, or a startup following a scram. This procedure may not be used for the initial approach to criticality with a new core or a revised core; see ANSOP 5723 or ANSOP 9 to cover these situations. This procedure shall not be used for recovery from a setback; see ANSOP 5701 for the setback recovery procedure.

PROCEDURE

1. Check that the pre-startup check list (ANSOP 5704) has been completed.

2. Set the alarm and scram points on the various circuits as follows:

   a. Main Loop Flow - set the scram contact at 85% of the value currently showing on FR-101; set the setback contact at 90% and the annunciator contact at 95%.

   b. Reactor Gas Outlet Temperature High - set the scram circuit at 1260°F; set the setback contact at 1240°F and the annunciator contact at 1220°F.

   c. Power Level High - set the scram contact at 115% of anticipated ion chamber current at the contemplated operating level. The anticipated current should be determined by the best means available (previous operating records, extrapolation of previous operating records, etc.) Set the setback contact at 110% and the annunciator contact at 105%.

   d. Log N-High - set the scram contact at 115% of anticipated ion chamber current at the contemplated operating level as described above.

   e. Period Short - set the scram contacts in the period amplifiers at 7 seconds.

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- set the scram circuit in the period recorder at 7 seconds
- set the setback circuit contact in the channel I and II period chassis at 10 seconds

3. Check the instrument racks for startup conditions:

a. All annunciators clear.
b. All recorders operating.
c. All instruments operating normally.

4. Check the console for startup condition:

a. Power on.
b. Interlock circuits made up.
c. All instruments operating normally.

5. Announce over the PA system that the reactor is about to be started up.

6. Energize the magnets by depressing the button on the console center strip. Observe that the "magnets ON" light is illuminated.

7. Establish and record the LCR base count. During the following period, the rods will be withdrawn in increments. Following each withdrawal, the count rate will be observed for one minute and recorded. This mode of operation will continue until a count rate forty times the base rate is attained, or until all rods except the shims and fine rod are out. If the count rate exceeds forty times the base rate prior to attaining this configuration, shut down and consult Day Supervision.

During any rod withdrawal, if the count rate doubles, execute ABLE. Repeat the procedure using smaller increments of rod withdrawal.

The interval between any two increments of rod withdrawal during this phase shall be at least one minute.

8. Move the rod selector switch to GANG I.

9. Actuate the GANG I switch to move the rods full out. Check the console lights to observe that the rods have functioned properly.

10. Observe and record the new count rate.

11. If the count rate doubled, ABLE. Repeat with half gang increments of withdrawal.

12. Move the rod selector switch to GANG II.

13. Actuate the GANG II switch to move the gang out for 3.75 minutes. Observe for proper function.

14. Repeat steps 10, 11.

15. Repeat steps 8, 9, 10 and 11 for rods in the following sequence:

Rod or Gang	Withdrawal Increment
Gang II	6.5 minutes Total
#3 SD	Full Out
#4 SD	Full Out
#5 SD	Full Out
#6 SD	Full Out
Setback	Full Out

CAUTION - In this sequence of rod withdrawal, all rods except the shims and fine rod must be out before a count rate forty times the base count rate (step 7) is attained. If this is not the case, shut down and consult Day Supervision.

16. At some point in the startup, the count rate system will jam because of excessively high count rate. At this point:

a. Operate the Shield Control switches on the console to move the shields down over the fission chambers.

b. Observe that the count rate has declined indicating that the shields performed satisfactorily.

c. Operate the "Channel 1 & 2" interlock key switch on the console to turn off the power to the LCR system. Observe that the "off" light comes on; if not, the parallel interlock in the Log N recorder is not tripped.

17. After properly positioning the rod selector switch, withdraw the fine rod in increments of 1 inch with a one minute wait between each withdrawal. Stop fine rod withdrawal at 5.5 inches out and proceed to withdraw shim rods one at a time.

18. During rod withdrawal, observe all nuclear instruments for evidence of criticality. Criticality will be indicated by a sustained period or a sustained rise on a power indicator.

19. After attaining criticality, adjust the fine rod to regulate the period to approximately 30 seconds. Operate the range change switch on the linear power circuit to keep the recorder on scale.

20. Level off the power rise at 150 watts and proceed with ANSOP 5706 - Increasing Power.

STANDARD OPERATING PROCEDURES

GAS COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: NORMAL OPERATION

SCOPE

This procedure outlines the general activities associated with the normal operation of the reactor. Since, obviously, no sequence can be assigned to the various activities, they are discussed by subject with the understanding that they will be performed as required.

PROCEDURE

1. Manpower

a. At least two qualified reactor operators shall be physically in the control room at all times when the console power is on.

b. One qualified reactor operator shall be seated at, and in control of, the reactor operating console at all times when the console power is on.

c. The shift supervisor and operating superintendent are considered to be qualified reactor operators for the purpose of this section.

2. Instrument Surveillance

a. The console operator shall maintain essentially continuous surveillance of the data displayed on the instruments on the console. In addition, he shall periodically survey those portions of the instrument racks which he can discern from his operating station (annunciator panels, trends on recorders, etc.)

b. The other control room operator shall periodically survey all instruments in the instrument racks.

c. Both operators shall complete the instrument log sheets prescribed for their operating post in ANSOP 5712 - Routine Reports.

3. Reactor Control

a. The console operator shall make such adjustments to the rods as are indicated by the various instruments to maintain the reactor in the prescribed mode of operation. Rod adjustments shall be made in accordance with the procedures outlined in ANSOP 5710-Rod Operability Check, ANSOP

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5719 - Operation of Fine Rod, or ANSOP 5722 - Interchange of Fine and Shim Rod.

b. The console operator shall call on the other operator or the shift supervisor for assistance in interpreting data or completing routine reports if required.

c. The console operator shall advise the shift supervisor immediately of any unusual conditions which develop or occur relative to the control of the reactor.

4. Facility Surveillance

a. The control room operator (other than the console operator) shall make periodic tours of the facility as outlined in ANSOP 5712-Routine Reports.

5. Initiation of Emergency Procedures

a. Whenever a scram or setback annunciator sounds,

(1) The console operator shall immediately press the manual scram or manual setback button, corresponding to the signal.

(2) The control room operator shall immediately obtain a copy of the indicated emergency procedure.

(3) The console operator shall notify the supervisor.

(4) Both operators shall execute the emergency procedure.

b. Whenever an annunciator other than a scram or setback annunciator sounds,

(1) The control room operator shall immediately obtain a copy of the indicated emergency procedure.

(2) Both operators shall execute the emergency procedure.

6. Miscellaneous Activities

a. The control room operator shall, as required, perform minor maintenance on the control room instruments (straighten charts, change charts, clean pens, etc.)

b. The control room operator on 4-12:30 shift shall, near the end of the shift, prepare a set of ANSOL (per ANSOP 5712-Routine Reports) for the oncoming 12-8:30 shift; including logging "hold-over" deactivations on ANSOL 5751.

STANDARD OPERATING PROCEDURES

GAS COOLED REACTOR EXPERIMENT-I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: PRE-START-UP CHECK LIST

SCOPE

This procedure lists the checks that must be made prior to the initiation of ANSOP 5702, Normal Start-up. The purpose of this check list is to provide in one place a status report of all the auxiliary services and support systems to establish that the facility is ready for reactor start-up.

PROCEDURE

1. Check that the utility water system is in normal operation.

   a. If start-up is required, perform the applicable procedures in ANSOP 61.

   b. Check that all utility water system annunciators are clear as follows:

      A-6 Utility Water System, Low Pressure
      A-9 Utility Water Storage, High Level
      A-5 Utility Water Storage, Low Level

   c. Check that the Well Pump (EP-150) is on auto operation from PS-307.

   d. Check that the Fire Pump (EP-153-1) is on auto operation from PS-302.

2. Check that the cooling water system is in normal operation.

   a. If start-up is required, perform the applicable procedures in ANSOP 62.

   b. Check that all cooling water system annunciators are clear as follows:

      A-16 Cooling Tower Basin, High Level
      D-8 Cooling Water System, Pressure Low
      A-15 Cooling Tower Basin, Low Level

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c. Check that one cooling water pump (EP-125-1 or EP-125-2) is operating and that the other is on auto operation from PS-301.

3. Check that the utility air system is in normal operation.

a. If start-up is required, perform the applicable procedures in ANSOP 63.

b. Check that all utility air system annunciators are clear as follows:

A-7 Utility Air System, Pressure Low

c. Check that the utility air compressor (EP-200) is on auto-operation from PS-304.

4. Check that the instrument air system is in normal operation.

a. If start-up is required, perform the applicable procedures in ANSOP 64.

b. Check that all instrument air system annunciators are clear as follows:

B-13 Instrument Air System, Pressure Low

c. Check that the instrument air compressor (EP-210) is on auto operation from PS-303.

5. Check that the fuel oil system is in normal operation.

a. If start-up is required, perform the applicable procedures in ANSOP 65.

b. Check that all fuel oil system annunciators are clear as follows:

A-4 Fuel Oil Pressure, Low
B-8 Fuel Oil Storage, Level Low

c. Check that one fuel oil pump (EP-233-1 or EP-233-2) is operating and that the other is on auto operation from PS-311.

6. Check that the LPG System is in normal operation.

   a. If start-up is required, perform the applicable procedures in ANSOP 66.

7. Check that the steam system is in normal operation.

   a. If start-up is required, perform the applicable procedures in ANSOP 67.

   b. Check that all steam system annunciators are cleared as follows:

      (1) A-11 Steam Boiler, Burner Failure
      (2) A-12 Boiler Day Tank, Level High

8. Check that the ventilation system is in normal operation:

   a. If start-up is required, refer to ANSOP 69.

   b. Check that the following units are in operation:

      (1) Control room air conditioner
      (2) Reactor area exhaust fan
      (3) Reactor area supply fans (2)

9. Check that the demineralized water system is in normal operation.

   a. If start-up is required, perform the applicable procedures in ANSOP 68.

   b. Check that all demineralized water system annunciators are clear as follows:

      A-14 Demineralizer (EP-80) - Conductivity High
      D-5 Primary Cooling Water- Flow Low
      D-12 Demineralized Water Surge Drum - Level Low
      D-16 Demineralized Water Surge Drum - Pressure Low

   c. Check that one demineralized water pump is operating (EP-75-1) or EP-75-2) and that the other is on auto operation from FE-104.

10. Check that the waste nitrogen system is in normal operation.

   a. If start-up is required, perform the applicable procedures in ANSOP 52.

b. Check that all waste nitrogen system annunciators are clear as follows:

D-9 Waste Nitrogen Storage - High Level

11. Check that the oxygen system is in normal operation.

a. If start-up is required, perform the applicable procedures in ANSOP 53.

b. Check that all oxygen system annunciators are clear as follows:

B-12 Oxygen Storage - Low Pressure

12. Check that the nitrogen storage is in normal operation.

a. If start-up is required, perform the applicable procedures in ANSOP 51.

b. Check that all nitrogen storage annunciators are clear as follows:

B-11 N_2 Supply - Low Pressure
B-4 Blowdown N_2 Supply - Low Pressure

13. Check that the pool water system is in normal operation.

a. If start-up is required, perform the applicable procedures in ANSOP 56.

b. Check that all pool water annunciators are clear as follows:

D-11 Reactor Pool - Level High
A-13 Demineralizer (EP-103) - Conductivity High
D-10 Reactor Pool - Level Low

c. Check that pool water circulating pump (EP-100) is running.

14. Check that all valves are set for normal operation and that general condition of locked area is satisfactory for start-up in accordance with start-up valve check list, ANSOP 5708.

15. Check that the main loop is pressurized in accordance with ANSOP 51.

a. Check that all main loop condition annunciators are clear as follows:

B-9 Main Loop Make-up Flow-High
B-16 Main Loop, Oxygen Content High
B-10 Main Loop, Humidity High
D-2 Main Loop, Pressure High
D-3 Main Loop, Pressure Low
E-4 Water in Main Loop

16. Check that the main compressor (EP-52) is operating normally.

a. If start-up is required, perform the applicable procedures in ANSOP 51.

b. Check that all main loop circulation annunciators are clear as follows:

B-14 Compressor Seal Water-Pressure Low
B-15 Compressor Seal Water or Lube Oil Reservoir-Low Level
D-1 Main Loop - Flow Low
B-6 Compressor Bearing Oil - Temperature High
B-5 Tach Speed Low

17. Check that the temperature monitoring system is in normal operation per ANSOP 5705 and that annunciators B-8 (Fuel Element High Temperature) and E-5 (Thermocouple Manifold Pressure High) are clear.

18. Check that MV-203 operates properly.

19. Check that the heater (EP-53) is in normal operation per ANSOP 51.

a. Check that heater annunciators and system temperature annunciators are clear as follows:

E-1 Heat Exchanger-Gas Outlet Temperature-High
D-4 Reactor Gas-Outlet Temperature-High
E-6 Inlet Duct-Temperature-High
E-7 Exhaust Duct-Temperature-High
D-6 Primary Cooling Water-Temperature-High
D-7 Secondary Heat Exchanger-Outlet Water Temperature-High
A-1 Heater Burner Failure
A-2 Heater Burner Failure
A-3 Heater Burner Failure

20. Check that safety circuit checks have been completed per ANSOP 5707.

21. Check that neutron detection system is operable per ANSOP 5709 and that annunciator E-3 (Nuclear Instrument Failure) has been cleared.

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22. Check that all rods are operable per ANSOP 5710 and that annunciators E-9 (Water in Rods) and A-10 (Safety Rods Low Pressure) have been cleared.

23. Check that the emergency generator (EP-290) is operating normally per ANSOP 5010 and that annunciator B-1 (Emergency Power Failure) is clear.

24. Check that the RAM system is operable per ANSOP 24 and indicates normal radiation levels. Check that annunciator E-2 "RAM - High Activity" is clear.

25. Check that the Fission Product Monitor is in normal operation per ANSOP 5713.

26. Check that the wind indicator is in normal operation.

27. Check that pre-start-up instrument check has been completed (ANSOL 5714).

STANDARD OPERATING PROCEDURES

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: INCREASING POWER

SCOPE

This procedure outlines the steps to be followed in increasing the power level of the reactor from 150 watts to any higher prescribed level. The procedure assumes that the 150 watts was attained under procedure ANSOP 5701 (Fast Start-up) or ANSOP 5702 (Normal Start-up).

The procedure shall also be used for increasing power at any power level if the increase is greater than 100 KW.

PROCEDURE

1. Prepare ANSOL 5753 to evaluate the effects of the contemplated power increase. If the evaluation is satisfactory, proceed with step 2. If the evaluation is not satisfactory, discuss with Day Supervision.

2. Check that PIC-102 is operating properly by momentarily increasing the set point to check operation of PCV-102-A and then returning set point to normal to check operation of PCV-102-C.

3. Check that TIC-105 is operating properly by momentarily increasing the set point to check operation of TCV-105. Return the set point to normal.

4. Check that the fine rod is on manual control. If not, switch to manual control.

5. During the power increase, the operator will observe all neutron instruments for evidence of unanticipated events. The primary control instrument will be TI-103 and the associated recorder TR-103 (Reactor Gas-Outlet Temperature).

6. Remove sufficient fine rod to establish a positive period as shown on the period indicator. At no time during the rise shall the period be shorter than 30 sec.

7. Adjust the fine rod to control the rate of power increase so that the reactor gas outlet temperature increases 15°F per minute. During the power

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increase operate the range switch on the linear power recorder to keep the trace or scale.

8. Do not withdraw the fine rod beyond 5 inches from full out. When the rod reaches this point, insert rod to increase the period to and hold the power level constant for at least five minutes.

9. Interchange the fine and shim rods per ANSOP 5722.

10. Continue power increase until the desired power level is reached.

11. Perform adjustments as required on the fine rod and switch the fine rod control to automatic.

12. Complete ANSOL 5753 and compare contemplated data with attained data.

STANDARD OPERATING PROCEDURES

GAS-COOLED REACTOR EXPERIMENT-I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: SAFETY CIRCUIT CHECK

SCOPE

This procedure outlines the check to be made of the safety circuits to establish that each element of the circuit is in proper operating condition. This check is required within four hours of the initiation of any start-up except the setback recovery (ANSOP 5701) or the recovery from a scram under ANSOP 5702.

The basic method used to check operability of safety circuits is to observe that the magnet current is interrupted when the actuating device in the circuit is tripped.

PROCEDURE

1. Check that all rods are in the reactor. This will energize K-98 relay and close the first of the interlocks.

2. Close the keyed by-pass switch around MS-49 to simulate the proper exit temperature on the fired heater.

3. Check that the reactor pool water level is normal. If so, K-99 will be picked up. If not, close the keyed by-pass switch to simulate proper pool water level.

4. Check that the demineralized water system is in normal operation. If so, K-100 will be picked up. If not, close the keyed by-pass switch to simulate proper operation.

5. Check that the cooling water system is in normal operation. If so, K-101 will be picked up. If not, close the keyed by-pass switch to simulate proper operation.

6. Check that the main loop pressure is normal and that MS-51 in the PR-102 is closed. If not, close the keyed by-pass switch to simulate proper pressure.

7. Check that the rod selector switch is on the "Off" position and hang a "Danger" tag on this switch.

8. Turn on the keyed main power switch. Observe that the indicating light in the power circuit comes on.

9. Check that the counting rate on the channel I and II log count rate recorder is sufficiently high to close MS-48 in this recorder.

If so, the interlock light on the center strip should be on, indicating that the interlock circuit is complete.

If not, depress MS-48 manually and observe that the interlock light comes on. Release MS-48. The light should stay on as K-92 seals in around the interlock circuit.

10. Open all keyed by-pass switches closed in steps 2, 3, 4, 5, and 6 above.

11. Close the following keyed bypass switches:

a. Main loop flow - around MS-50 in FR-101.

b. Log N - around MS-53 in Log-N recorder.

c. Period - around MS-52 in Channel I, II, III or IV period recorder.

12. Check that all reactor room doors are closed. Doors which are interlocked are listed below:

Door No.	Function
29	Corridor to Contaminated Work Area
31	Corridor to Wash Down Room
36	Mech. Equipment Room to Outside
40	Work Area to Outside
39	Work Area to Outside
32	Wash Down Room to Outside

13. Check that the drive pressure on the safety rods is at 125 psi and that the mercoid is closed.

14. Check that area run-safe switches are on run. Switches are located as follows:

South wall of Heater Pit

East wall of Mechanical Loop Pit

East wall of Mechanical Equipment Room

South wall of Work Area

15. Check that the safety amplifiers are in normal operating condition.

16. Turn the run-safe switch on the console to run.

17. Depress the "Magnet Energize" button on the console and observe that "Magnet ON" light comes on.

18. From this point, two operators are required to perform the check. Communication between the two is required.

19. Operator One - manually, hold down MS-50 in FR-101
Operator Two - open the keyed by-pass on main loop flow
Operator One - release MS-50
Operator Two - Observe that the "Magnet ON" light goes out. Close the keyed by-pass switch and re-energize the magnets.

20. Operator One - manually hold down MS-53 in the Log N recorder.
Operator Two - open the keyed by-pass switch on Log N.
Operator One - release MS-53.
Operator Two - observe that the "Magnet ON" light goes out. Close the by-pass switch and re-energize the magnets.

21. Operator One - manually hold down MS-52 in the period recorder.
Operator Two - open the keyed bypass switch on Period.
Operator One - release MS-52.
Operator Two - observe that the "Magnet ON" light goes out. Close the by-pass switch and re-energize the magnets.

22. Operator One - obtain an SWP and proceed on a tour of the area. Trip the circuits listed below. Call in to control room on the intercom to verify results.
Operator Two - Observe that the "Magnet ON" light goes out each time a circuit is tripped. Re-energize magnets after each trip.

a. Door 29 Interlock - open and close door, enter area, close door.

b. Earthquake switch - open door on switch box and, with a pencil, make the contact points touch. Close door.

c. Safety rod low pressure - remove mercoid switch cover and trip mercury bottle. Reset and replace cover.

d. Run-safe switch - heater pit wall. Turn to safe, then back to run. At each run-safe switch, observe that the indicating lights function properly.

e. Run-safe switch - mechanical loop pit. Turn to safe, then back to run.

f. Run-safe switch - work area. Turn to safe, then back to run.

g. Door 40 in work area - open and close.

h. Door 39 in work area - open and close.

i. Run-safe switch in Mechanical Equipment Room. Turn to safe, then back to run.

j. Door 36 in Mechanical Equipment Room - open and close.

k. Door 32 in Wash area - open and close.

l. Door 31 in Wash area - open, enter corridor, then close.

m. Return to the control room via the change room. Terminate the SWP.

23. Operator One - open MS-52 in recorder 103 (Reactor Outlet Gas Temperature.)
Operator Two - observe that "Magnet ON" light goes out.
Operator One - release switch.
Operator Two - re-energize magnets.

24. Operator One - using special signal generator and Test Procedure for Safety Amplifier (ANSOP 8_____), trip each safety amplifier in turn on high level and short period.
Operator Two - on each trip, observe that "Magnet ON" light goes out and re-energize magnets.

25. Un-bypass the circuits by-passed in step 11 above.

26. Operator One - locate and observe the action of relay K-77 in Rack #1.
Operator Two - actuate the following circuits, checking with Operator One that K-77 closes as each circuit closes.

a. Manual setback button
b. MS-45 in Main loop flow recorder 101
c. MS-46 in Linear power recorder R-10
d. MS-47 in Reactor outlet temperature recorder 103
e. With special signal generator and test procedure for low level period amplifier, trip the period circuit in channel I and channel II amplifiers.

27. Secure relay K-77.

28. Open the main power switch on the console.

29. Remove the "Danger" tag from the rod selector switch.

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STANDARD OPERATING PROCEDURES

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: START-UP VALVE CHECK LIST

SCOPE

This procedure outlines the checks to be made on the main loop system to assure that it is in condition for reactor operation. Generally, the valves will have been placed in the proper condition by other procedures covering the start-up of the individual components. However, this last check will assure that the system is ready and will uncover any errors in the earlier work.

PROCEDURE

An operator is to personally check each of the following valves to be sure it is in the indicated position. Use ANSOL 5707 as a check list. During the tour of the area, the operator will make a visual check of the condition of the locked area to determine that the general condition of the equipment is satisfactory for start-up.

Valve No.	Location	Position
GNL-V-1	GNL-1 Loop Pit	Open
GNL-V-2	GNL-4 Loop Pit	Open
GNL-V-3	GNL-5 Loop Pit (Vent)	Closed
GNL-V-4	GNL-1 Loop Pit (PR-102)	Open
GNL-V-5	GNL-1 Loop Pit (Vent)	Closed
GNL-V-7	GNL-2 Loop Pit (Vent)	Closed
GNL-V-8	GNL-2 Loop Pit (HIC-OIC)	Open
GNL-V-9	GNL-2 Loop Pit (PG-101)	Open
GNL-V-10	GNL-3 Loop Pit (PG-102)	Open
GNL-V-11	GNL-3 Loop Pit (Vent)	Closed
GNL-V-12	GNL-4 Loop Pit (Vent)	Closed
GNL-V-13	GNL-4 Loop Pit (PR-102)	Open
GNL-V-15	GNL-4 Loop Pit (LI-102)	Closed
GNL-V-14	GNL-4 Loop Pit (Vent)	Closed
GNL-V-50	GNL-4 Loop Pit (Purge)	Closed
GNL-V-49	GNL-20 Loop Pit (Flood)	Closed (Locked)
GNL-V-16	GNL-8 Loop Pit (Filter Loop)	Open
GNL-V-17 or 18	GNL-8 Loop Pit (Filter Loop)	Open*/Closed
GNL-V-19 or 20	GNL-9 Loop Pit (Filter Loop)	Open*/Closed
GNL-V-21	GNL-17 Loop Pit (Filter Drain)	Closed

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Valve No.	Location	Position
GNL-V-22	GNL-17 Loop Pit (Filter Drain)	Closed
GNL-V-23	GNL-8 Loop Pit (PdI-108)	Open
GNL-V-24	GNL-9 Loop Pit (PdI-108)	Open
GNL-V-25	GNL-10 Loop Pit (Filter Loop)	Open
GNL-V-40	GNL-12 (Make-up)	Open
GNL-V-41	GNL-12 Loop Pit (Make-up)	Open
GNL-V-39	GNL-16 Loop Pit (Make-up)	Open
WA-V-6	GNL-1 & WA-1 Loop Pit	Closed (Locked)
CS-V-5	GNL-1 & CS-2 Loop Pit	Closed (Locked)
CS-V-6	GNL-4 & CS-3 Loop Pit	Closed (Locked)
WA-V-7	GNL-4 & WA-2 Loop Pit	Closed (Locked)
GNL-MV-105	GNL-2 Loop Pit (Blowdown)	Closed
GNL-MV-106	GNL-2 Loop Pit (Blowdown)	Closed
GNL-MV-107	GNL-4 Loop Pit (Blowdown)	Closed
GNL-MV-108	GNL-4 Loop Pit (Blowdown)	Closed
GNL-MV-103	GNL-6 Loop Pit (Emerg. Cooling)	Closed
GNL-MV-104	GNL-7 Loop Pit (Emerg. Cooling)	Closed
S-MV-110	GNL-4 & S-3 Loop Pit	Closed
WP-V-19	WP-13 Loop Pit (Flooding)	Closed (Locked)
WP-V-21	WP-12 Loop Pit (Flooding)	Closed (Locked)
WP-V-22	WP-12 Loop Pit (Flooding)	Closed (Locked)
WP-V-21	WP-12 Loop Pit (Flooding)	Closed (Locked)

*One or the other of these valves will be open, the other will be closed.

STANDARD OPERATING PROCEDURES

GAS COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: RULES FOR DEACTIVATION OF INTERLOCK AND SAFETY CIRCUITS

SCOPE

This procedure outlines the rules governing deactivation of interlock and safety circuits and establishes the procedures under which such deactivations may be accomplished.

It is the primary purpose of this procedure to prevent a situation where an interlock or safety circuit is unknowingly disabled or knowingly deactivated without proper consideration for the consequences. ANTS-105 provides the basic limitations on the latter phase.

PROCEDURE

A. Wiring Changes

1. Wiring changes in the control room or the portion of the control circuitry outside the control room will be authorized by the Operating Superintendent on form ANSOL 5051 - Wiring Alteration Authorization.

2. Jumpers shall be considered to be wiring changes and shall be subject to the above restriction.

B. Lock Switches

1. Keys to the interlock and safety circuit lock switches shall be identified with red plastic tags approximately 2" in diameter. The tag will identify the key by number.

2. Red tag keys will be stored in a locked key cabinet with an individual hook for each key. Access to this key cabinet shall be limited to the Operating Superintendent and the Shift Supervisors.

3. Red tag keys shall be inventoried and accounted for at the end of each shift. The Shift Supervisors Log (ANSOL 5799) shall indicate that the inventory is complete.

4. Keys to valve and equipment operating lock switches shall be identified with a clear plastic tag. These keys shall be stored on a key board and are available for use as required.

C. Deactivations

1. Deactivations for safety circuit checks, with all rods in the reactor may be made without approval as prescribed in the procedure, ANSOP 5707.

2. Deactivations for rod operability check may be made without approval as prescribed in the procedure ANSOP 5710.

3. Deactivations made for any purpose while the main power on the console is "off" may be made without approval as required.

4. Deactivation of any interlock circuit may not be affected between the time console power is turned on for a start-up and the time magnets are energized. These interlocks may be deactivated at any other time as required.

5. All other deactivations shall be made only with the approvals indicated and for the time specified on the below table. All such deactivations shall be logged on ANSOL 5751 prior to the deactivation.

Circuit	Approval Required	Max. Time Deactivated
Manual scram	---	0
Power failure-scram	Operating Supt.	2-hours
Gas flow low-scram	---	0
Gas temperature high-scram	---	0
Power level high	---	0
Seismometer-scram	Operating Supt.	2-hours
Reactor room doors-scram (1)	(Shift Supervisor	15-minutes
	(Operating Supt.	4-hours
Run safe switch-scram (1)	Shift Supervisor	4-hours
Log N high-scram	Shift Supervisor	2-hours
Period short, channels I, II, III or IV-scram	Shift Supervisor	2-hours
Period short, channels III and IV-scram	---	0
Safety rod pressure low-scram	---	0
Manual setback	---	0
Period short-channels I and II setback	Shift Supervisor	See Note (2)
Gas temperature-high setback	Shift Supervisor	See Note (3)
Gas flow low - setback	Shift Supervisor	See Note (3)
Power level high-setback	Shift Supervisor	See Note (3)

Notes: (1) These two circuits may not be deactivated at the same time.
(2) Indefinitely when power level is greater than 150 watts.
(3) Indefinitely for maintenance. Must be activated when operable.

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STANDARD OPERATING PROCEDURE

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 - REACTOR OPERATION

TITLE: OPERATION OF FISSION PRODUCT MONITOR

SCOPE

This procedure outlines the method of operation of the fission product monitor.

Basically this device samples the gas in the main loop for evidence of fission fragments which would indicate a failure of the cladding on a fuel element. Once a failure is detected, the monitor is capable of isolating the fuel element in which the failure exists. Some special techniques of main loop operation are required for this latter phase.

In the event that the observation of any particular energy range is desired the pulse height analyzer can be adjusted to present the data.

PROCEDURE

A. Start-up of Fission Product Monitor (FPM)

1. Turn on the master FPM power switch (S-53) on rack 15, panel 2.

2. Turn on the power switches for linear and log ratemeter, pulse height analyzer, and high voltage supply; panels 2, 3, 4, and 6, rack 16.

3. Check the valve rack in the reactor area as follows:

a. Main manifold purge valve closed. (FPM-V-3)

b. One loop sampling line block valve open and the other closed. (FPM-V-1 and FPM-V-2).

c. Block valves in all individual fuel element sampling lines open. (FPM-V-A-1 through A-5, FPM-V-B-1 through B8, FPM-V-C-1 through C-9, FMP-V-D-1 through D-10, FPM-V-E-1 through E-9, FPM-V-F-1 through F-10, FPM-V-G-1 through G-9, FPM-V-H-1 through H-8, FPM-V-I-1 through I-5).

d. Both block valve in the manifold outlet line open (FPM-V-4 and FPM-V-5).

4. Check that the main loop is pressurized to at least 50 psig.

5. Check that the main loop flow is at least 5 pounds per second.

6. For this step, two operators are required.

a. Operator I - check that channel selector switch (S-54) on rack 16, panel 3, is in "Monitor" position.

b. Operator II - observe that the flow meter (FE-450) and the high and low pressure gauges (PG 452 and 453) on the FPM rack in the reactor area are indicating a gas flow.

c. Operator I - move the selector switch from "Monitor" to "Channel A" and the fuel element switch from "Off" to "1". Observe that the chart speed on the recorder, rack 16, panel 1, increases.

1) In the event that the chart speed does not increase when changing from "Monitor" to "Fuel Element Selector", actuate relay K-117 on the back of panel 3 rack 16, manually. If this action does not increase the chart speed, maintenance is required.

d. Operator II - observe that the flow rate (FE-450) has doubled. If not, advise Operator I.

e. Operator I - if the change in the flow rate does not occur, contact the Shift Supervisor.

f. Operator I - select each fuel element in turn by operating the "Channel Selector Switch" and "Fuel Element Selector Switch" on panel 16, rack 3. Sufficient time must elapse between selection to permit the high pressure gas trapped between the block valve and the regulator to bleed off through the regulator.

g. Operator II - as the fuel elements are selected, observe the high and low pressure gauges (PG 452 and 453). All high pressure readings should be the same as the main loop pressure, within 10%. All low pressure readings should be 30 - 40 psig. Report any reading outside this limit to the control room.

h. Operator I - as the fuel elements are selected, check the pressure indication on the meter on rack 16, panel 3. All readings should be 30 to 40 psig.

i. Note any reading outside of limits and report these along with any reported by Operator II in step "g" above to the Shift Supervisor.

j. Return the channel selector switch to "Monitor" position and fuel element detector switch to "Off".

7. a. Operator II - place the calibration source adjacent to the detector assembly in the pit under the FPM valve rack.

b. Operator I - Attach an oscilloscope to the "Calibrate" connection on the pulses height analyzer chassis, rack 16, panel 4.

(1) Adjust the high voltage to 1000 volts.

(2) Set the analyzer gain to give 1 mev energy at 100 volts pulse height.

(3) Check the FPM count rate meters panel 2, rack 16, and associated recorder rack 16, panel 1, for significant and similar indications on both instruments.

c. Operator II - return the source to the storage location.

8. Leave the system in the following condition:

a. Channel selector switch (S-54) on "Monitor".

b. Fuel element selector switch (S-60) on "Off".

c. Fission product monitor master power switch (S-53) on "On".

B. NORMAL OPERATION

1. During normal operation, the control room operator will complete the Fission Product Monitor Log Sheet (ANSOL 5760) once every two hours.

2. Concurrent with (1) above, make the adjustments to the controls indicated below as required to conform to normal operating settings shown on the log sheet.

a. Pulse height analyzer high voltage (1000 volts)
b. Pulse height analyzer amplifier gain (1 Mev @ 100 volts)
c. Linear and log rate meter (approximately mid-scale)
d. FIC-108 (no flow).

C. PROCEDURE FOR MONITORING INDIVIDUAL FUEL ELEMENTS

1. Check that reactor is shutdown.

2. Advise console operator to decrease the set-point of PIC-102 to zero and shutdown the main compressor. GNL-PCV-102C will allow open to exhaust system nitrogen gas to nitrogen waste tank.

3. Concurrent with this operation, increase the set-point of FIC-108 to about 70% of maximum flow rate.

4. Advise the console operator to adjust TIC-104 to hold FH-101 outlet gas temperature less than 1100°F and TIC-101 to maintain reactor inlet gas temperature at 800°F.

5. During this operation continuous surveillance of the fuel element temperatures (TI-401 and all recorders) is required. If any fuel element temperature reaches 1600°F increase FIC-108 set-point to reduce the temperature. If this does not correct the condition, discontinue the operation and advise the console operator to institute an emergency blowdown.

6. Move the channel selector switch from "Monitor" to "Channel A".

7. Move the fuel element selector switch from "Off" to "1".

8. Observe the count rate recorder; it should be on fast chart speed and making a trace upscale then leveling off.

9. After about 10 seconds or after the trace is level for a discernable period of time, switch the fuel element selector switch from "1" to "2".

10. Repeat steps 7, 8, and 9 until position 17 is reached, then return fuel element selector switch to zero.

11. Move channel selector switch to the next channel.

12. Repeat steps 7, 8, 9, 10 and 11 until all fuel elements have been monitored.

13. Advise the console operator to increase the set-point of PIC-102 and startup the main compressor. (Note: Increasing PIC-102 set-point will close GNL-PCV-102-C.)

14. Concurrent with this operation decrease the set-point of FIC-108 slowly.

15. During this operation continuous surveillance of fuel element temperatures (TI-401 and all fuel element temperature recorders) is required. If fuel element temperatures reach 1600:

a. Open FIC-108 - wide.

b. If this does not correct this condition, discontinue the operation and advise the console operator to institute an emergency blow-down.

16. Continue this mode of operation till main loop flow rate is at normal rate and FIC-108 is completely closed.

17. Analyze the data from the recorder and identify the damaged fuel element.

D. PROCEDURE FOR OBSERVING DISCRETE ENERGY RANGE

1. Attach the oscilloscope to the calibrate position on the front of the analyzer.

2. Knowing the energy level of the product being observed, set the threshold voltage of the analyzer just below the energy peak, i.e., 20 small divisions of the dial. Set "window" in similar fashion above the energy peak. This procedure excludes all energy indications above and below the peak being observed.

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STANDARD OPERATING PROCEDURES

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: PRE-START-UP INSTRUMENT CHECK LIST

SCOPE

This procedure outlines the checks to be made of the control room instruments to assure that the support systems are in condition for reactor start-up. This check will be performed in sequence as shown in ANSOP 5704 - Pre-Start-up Check List.

PROCEDURE

1. Cooling Water System

a. Check TI-107; should be about 100°F. (Water to water exchanger outlet).

b. Check TI-302; should be above 40°F and less than 75°F. (Cooling tower basin).

2. Demineralized Water System

a. Check TI-110; should be about 110°F. (Gas to water exchanger inlet).

b. Check TI-108; should be about 125°F. (Gas to water exchanger outlet).

c. Check FI-104; should be about 1000 gpm. (Secondary loop flow.

3. Main Loop

a. Check FI-101; should be about 18 pounds per second. (Main loop flow).

b. Check TIC-101; should be about 800°F. (Reactor inlet).

c. Check TI-103; should be about 800°F. (Reactor outlet)

d. Check TIC-105; should be about 140°F. (Filter loop inlet)

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e. Check TIC-104; should be about 800°F. (Heater outlet)

f. Check PI-104; should be about 198 psi. (Compressor suction).

g. Check PI-106; should be about 240 psi. (Compressor discharge).

h. Check PIC-102; should be about 200 psi. (Main loop pressure).

i. Check PdI-102; should be about 13 pounds. (Reactor ΔP)

j. Check OI-101; should be less than 0.6% and greater than 0.3%. (Loop oxygen)

k. Check HI-101; should be less than 0°F. (Loop humidity)

l. Check FI-102; should be about zero. (Loop make-up).

m. Check FIC-105; should be about 110 cfm. (Filter loop flow).

4. Main Compressor

a. Check FIC-103; should be about 1.3 cfm. (Seal bleed)

b. Check PI-107; should be about 300 psig. (Seal water)

c. Check TACH-102; should be about 8730 rpm. (Compressor speed)

5. Neutron Panel

a. Check LCR recorder on both channels; should show significant count rate.

b. Check Log-N recorder on both channels; should show zero.

c. Check period recorder on all four channels; should show ∞

d. Check safety recorder on all four channels; should show zero.

e. Check linear power recorder; should show zero.

6. Reactor Pool

a. Check TI-201; should be less than 110°F. (Pool water).

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STANDARD OPERATING PROCEDURE

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION : 5 - REACTOR OPERATION

TITLE: REACTOR SHUTDOWN

SCOPE

This procedure outlines the steps to be followed in effecting a normal shutdown of the reactor. This procedure will be used for shutdowns for fuel element handling, reactor or loop maintenance work, or in the execution of any emergency procedures which require a normal shutdown.

The reactor proper may be shutdown by four other means; automatic scram, automatic setback, ABLE and BAKER (see ANSOP 5801). If the reactor is to be restarted in the immediate future (i.e., the cause of the shutdown is known and can be corrected quickly and the plan of operation requires a restart), no action subsequent to the automatic action or that outlined in ANSOP 5801 is required. If the reactor is to be left in a shutdown condition for some time, any of the above four shutdowns should be followed by the steps beginning with step 3 of this procedure.

The reactor shutdown specified in this procedure is the fastest consistent with no thermal shock to the unit. The following steps prescribe the order for shutting down the support facilities to effect a complete shutdown of the main system.

PROCEDURE

1. Set the fine rod on "Manual".

2. Insert fine rod to reduce the rate of reaction. Observe TI-103 and control the rate of temperature drop at about 15°F per minute.

3. If the main loop is to be depressurized, follow the system pressure down by adjusting the PIC-102 control point to avoid any make-up or bleed off of gas.

4. When the power level is less than five percent of the original indicated level as indicated by BTUI-101 and the nuclear instruments, run all rods in as follows:

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a. Gang II and the fine rod simultaneously

b. Shim gangs

c. Set-back rod

d. Gang I

5. Concurrent with step 4 reduce the heater outlet temperature 15°F per minute by adjusting the TIC-104 control point.

6. Maintain shutdown cooling in accordance with ANSOP 5717, using the main compressor if possible.

7. Proceed to prepare the loop or auxiliary equipment for repair, etc., as indicated in the shutdown schedule or emergency procedure.

STANDARD OPERATING PROCEDURE

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 - REACTOR OPERATION

TITLE: SHUTDOWN NEUTRON MONITORING

SCOPE

This procedure outlines the activity associated with the neutron monitoring of the reactor during shutdown conditions. The provisions of this procedure apply at all times when there is fuel in the reactor and the neutron detection system is not under surveillance as a part of another procedure (Start-up, Flooding, etc.).

The shutdown monitoring requirements are divided into three portions to cover the below described situations:

A. Reactor shutdown with no work in the reactor vessel (cover not removed),

B. Reactor shutdown with fuel being changed (no increase in core size or loading).

C. Reactor shutdown with fuel being added to the core (additional elements or more heavily loaded elements).

PROCEDURE

A. NORMAL SHUTDOWN

A-1 Activate either Channel I or Channel II per ANSOP 5709. The recorder must be operable and running.

A-2 A qualified reactor operator must be present in the control room at all times during reactor shutdown.

A-3 The recorder shall be observed periodically and any increase observed carefully.

A-4 Once per hour, the recorder case shall be opened and a tick mark made on the trace. This mark shall be identified with the time and the operator's initials.

An increase of a factor of 1.5 shall be reported immediately to the Shift Supervisor. Such an increase shall cause the initiation of an immediate investigation to determine the cause. When such an increase is noted, continuous surveillance of the system shall be initiated and such surveillance shall continue until the cause is established and/or no further change is noted for fifteen minutes.

B. NORMAL CHARGE - DISCHARGE OPERATIONS

B-1 Follow steps A-1 and A-2 above.

B-2 When advised the fuel element handling is about to begin (ANSOP 5721, step 5), begin continuous surveillance on the LCR recorder.

Open the case and mark the trace each 15 minutes, indicating time and initials.

An increase by a factor of 1.5 will require the following action:

a. Notify the fuel handlers to discontinue operations.
b. Notify the supervisor immediately.
c. No further fuel element handling until the cause of the increase is established and/or no further change in LCR is noted for 15 minutes.

C. ADDITION OF FUEL

C-1 Follow steps B-1 and B-2 above except that, after each element is added, observe the LCR for five minutes.

C-2 If the LCR remains constant or if the LCR increases by less than a factor of 1.5 and levels out, advise the fuel handlers to proceed.

C-3 If the LCR increases by more than a factor of 1.5 or does not level out within two minutes, advise the fuel handlers to remove the element. Consult with supervision before proceeding.

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STANDARD OPERATING PROCEDURES

GAS-COOLED REACTOR EXPERIMENT-I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: REACTOR COOLING FOLLOWING REACTOR SHUTDOWN

SCOPE

This procedure outlines the several methods of reactor cooling following a reactor shutdown. Since most of the emergency operations are automatic, the operators responsibility is largely one of checking to see that the proper sequence of events is occurring.

The procedure is divided into separate sections covering the various combinations of conditions which might exist following a reactor shutdown.

A. Normal shutdown cooling
B. Emergency shutdown cooling
C. Blowdown system cooling
D. Steam cooling
E. Shutdown cooling loop operation

Normally shutdown cooling will include Operations A and E only. Operation B is a back-up for A. Operation C is an emergency operation in case A and B are not operable and is normally followed by E. D is a final emergency procedure in case of failure or inoperability of A, B and C. It also is normally followed by E.

PROCEDURE

A. Normal Shutdown Cooling

This term defines the situation where the reactor has been shutdown by normal or emergency (either manual or automatic) procedure with all other components in the system intact and operable. In this case, the main compressor will continue to operate and circulate gas through the reactor and gas heat exchanger. The heater must be tripped off and the loop pressure control set back. The gas heat exchanger will automatically by-pass demineralized water to control the gas outlet temperature at 140°F.

1. Check to see that the main compressor set is still operating on the main motor. This is determined by observing that TACH 102 is reading greater than 4800 rpm.

2. Trip off the fired heater by setting the control point on TIC-104 to zero.

3. Close GNL-TCV-101A and open GNL-TCV-101B to by-pass the heater. This is accomplished by adjusting the set-point on TIC-101 over a 15 minute period to avoid temperature peaking in the heater tubes from stored energy in the fire brick.

4. As the system cools off, observe and adjust the following instruments:

a. PIC-102 (Main Loop Pressure). As the temperature drops, the pressure will also drop. In order to avoid adding gas to the system which will have to be exhausted during the subsequent startup the operator will observe the output signal indicator on PIC-102 and adjust the pressure set point to maintain the output signal at zero.

b. FI-101 (Main Loop Flow). As the temperature and pressure drop, with the volume being held essentially constant, the flow will change slightly as a result of altered system conditions. The operator will observe FI-101 and adjust the compressor speed control to maintain constant mass flow rate in the system.

c. TI-103 (Reactor Gas Outlet Temperature). This reading should decrease gradually and steadily as the residual heat in the heater is extracted along with the delayed neutron and gamma heat in the reactor and dumped in the gas heat exchanger. No adjustment is required but if the temperature increases, proceed immediately to step 5.

d. TI-105 (Gas Heat Exchanger Gas Outlet Temperature) and TI-106 (Main Compressor Discharge Gas Temperature) shall be observed to see that they decrease smoothly and steadily.

e. TdI-102 (Reactor Differential Temperature) shall be observed to see that it remains essentially constant after the initial drop resulting from the shutdown.

f. TI-401 (Fuel Element Temperature) shall be observed continuously. Any deviation from a steady gradual drop, i.e. if the temperature increases, proceed immediately to step 5.

5. If the temperature rises:

a. Check FI-104 (Demineralized Water Flow) and operating controls for demineralized water pumps. Make necessary adjustments if flow is low.

b. Check FI-101 (Main Gas Loop Flow) and adjust compressor speed if flow is low.

c. If neither of the above conditions is abnormal or if corrective action taken above does not reduce the temperature, adjust PIC-102 to pressurize the system.

d. During pressurization, control compressor speed to regulate flow.

e. When temperatures are again dropping, continue with step 4 of this procedure.

B. Emergency Shutdown Cooling

This term defines the situation where the reactor has been shutdown by a scram and the main compressor drive motor has also shutdown. In this condition, the eddy current coupling will automatically uncouple the main drive motor and the emergency drive motor on the main compressor will start automatically. This arrangement will circulate about half the normal volume of gas in the loop. The condition will be signaled by a 4620 rpm reading on TACH 102. The cooling will follow the same pattern as outlined in Section "A" above except that the temperature will decrease at a slower rate.

Follow steps 2, 3, 4 & 5 under Section A.

C. Blowdown System Cooling

Note: This system is an emergency device and shall not be instituted except in an emergency condition.

This term defines the condition where the reactor shutdown occurs simultaneously with a failure of both the main drive motor and the emergency drive motor of the main compressor set, or a seizure of bearings or some other condition which causes a loss of rotation of the main compressor. It is signalled by a zero reading on TACH 102 and a zero reading on FI-101. Such a condition will initiate a low flow scram and a low TACH indication. If these two conditions are followed by a fuel element temperature high alarm, the blowdown will be initiated automatically. Good operating practice dictates that a manual initiation of blowdown cooling should be effected to avoid high fuel element temperatures.

1. Determine that TACH 102 is less than 4620 rpm and dropping and that FI-101 is dropping.

2. Actuate the emergency blowdown system by pressing the button on the left console panel. Observe that the operating lights for MV-105, MV-106, and MV-108 are lit. If not operate the individual valve controls to open these valves.

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This will open MV-105 and MV-106 to bleed down the loop to the waste gas holder. Since these valves are on the compressor suction line, the check valve in the compressor discharge will cause the depressurization to sweep gas through the reactor.

Simultaneously MV-107 and MV-108 will open admitting 300 psig nitrogen from the emergency blowdown storage bank. A restrictive orifice (106) in the line limits the flow to about 3000 lbs/hour at 10 psig. This gas will sweep through the reactor and exit the system through MV-105 and MV-106 to the waste gas holder.

3. Trip off the heater by setting the control point on TIC-104 to zero.

4. Observe and adjust the system instrumentation as follows:

a. PIC-102 (Main loop pressure) should be set to zero to prevent any operation of normal make-up and bleed valves. The pressure will drop to 10 psig and level at this point.

b. FI-101 (Main loop flow) will drop to zero and remain there.

c. All temperature instrumentation will show a gradual and steady drop.

5. Dispatch an operator to the gas storage platform to check that all bottles are valved into the manifold and to operate GNL-V-42 if so instructed by the control room to valve the normal make-up bank into the blowdown system.

6. Observe the level in the waste gas holder on LI-201. When the annunciator (D-9) sounds, the tank is 75% filled and blowdown may continue for only 4 more minutes. The total gas volume of the holder is about 800 lbs. If any significant quantity of gas was in the holder when the blowdown started, action must be taken to blow this or the blowdown gas to the stack. The stack blower will handle about 800 lbs. with a 10:1 dilution in 15 minutes.

7. Unless initiated manually, the shutdown Cooling Loop will automatically go into operation when the pressure on the blowdown supply header reaches 290 psig, and the loop pressure reaches 10 psig.

D. Steam Cooling

This term defines the emergency situation where no circulation is possible with the main compressor and where either the shutdown cooling loop is inoperable or where gas temperatures are too high to permit its use. In

this case, reactor cooling is effected by blowing steam through the core.

1. Check that the main drive motor and emergency drive motor on the main compressor are inoperative. (Tach 102 zero).

2. Check that the shutdown cooling loop is:

a - inoperative or

b - cannot be used because TI-103 indicates greater than 1200°F.

3. Dispatch an operator to close main loop valves GNL-V-1 and GNL-V-2.

4. Open valves S-MV-110 and S-MV-201 using controls on Rack 3. Observe indicating lights to check that valves have opened.

5. Dispatch an operator to check that cooling water valves WC-V-8 is open and WC-V-7 is closed on the steam condenser.

6. Observe the fuel element temperature indicator (FI-401) and the system temperature indicator (TI-103). These should both be dropping.

7. Discontinue steam cooling and activate shutdown cooling loop when TI-103 indicates less than 1200°F. Before starting the shutdown cooling loop, add nitrogen to the system through GNL-V-50 until PIC-102 shows 5 psig.

E. Shutdown Cooling Loop Operation

This term defines the situation where reactor shutdown cooling is being accomplished by circulating the gas through the air cooled shutdown cooler with the shutdown cooling compressor. This is a low pressure (max. 10 psig) system with a capacity of 2000 lbs/hr at 10 psig.

The air cooled exchanger is designed for an inlet temperature of 1200°F but will normally operate with about 450°F maximum inlet to deliver 150°F outlet gas. The compressor is designed for a maximum of 450°F inlet gas conditions.

Remote operated valves in the loop are closed during normal operation. When the Shutdown Cooling Loop is activated, either automatically by the blowdown circuitry or manually from the console, both remote operated valves open, the compressor drive motor starts and the cooler fan starts.

1. During the preceding cooling operation, observe the reactor gas exit temperature on TI-103. As soon as this temperature reaches 1200°F, check the system pressure on PIC-102. If it is 10 psig or less, the shutdown

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Cooling Loop may be started by operating the toggle switch on the left console panel.

Note: If Shutdown Cooling Loop operation is being initiated out of sequence, check that the main compressor is shut down before initiating this step.

2. Immediately close MV-107, MV-108, MV-105 and MV-106 in that order. If they are open observe the indicating lights to check that the valves have closed.

3. Observe that the shutdown Cooling Loop has gone into proper operation by checking indicating lights on the console:

MV-103 -	Open
MV-104 -	Open
Compressor -	Running
Cooler Fan -	Running
FI-107 -	Near full scale

4. Observe the temperature and pressure instruments to see that the cooling is proceeding normally.

a. TI-111 (Shutdown Cooling Loop - Compressor Inlet Temperature). This should fall steadily. If it exceeds 450°F, shut down and initiate emergency blowdown (Section C above).

b. FI-107 (Shutdown Cooling Loop-Flow). This should hold relatively steady at a level between 1200 and 2000 lbs/hr depending on system pressure.

c. PIC-102 (System Pressure) Observe this instrument closely. If the pressure rises above 10 psig immediately open MV-105 to reduce pressure to less than 10 psig.

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STANDARD OPERATING PROCEDURE

GAS-COOLED REACTOR EXPERIMENT-I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: REACTOR FLOODING

SCOPE

This procedure outlines the steps to be followed in flooding the reactor. Prime considerations in the procedure are:

1. Control of water addition rate to prevent an approach to criticality.

2. Control of fuel element temperatures to avoid excessive thermal shock and steam generation.

In general this procedure is performed only when access to the reactor interior is required to change fuel elements or investigate internal equipment. It is not contemplated that the procedure will ever be used under emergency conditions; i.e., reactor flooding will always be scheduled.

The procedure is executed under the direct guidance of the shift supervisor, who directs the operation from the control room, where all data are available.

PROCEDURE

1. Check that the reactor is shut down and that the main power switch is off on the console.

2. Make a visual check to determine that all rods are in the reactor.

3. Check that neutron detector channels I and II are operative per ANSOP 5709 and establish continuous surveillance on the LCR indicator.

4. Check that shutdown cooling has been and is being provided per ANSOP 5717 and that fuel element temperatures are generally in the range of 200°F.

5. Fill EP 253 flooding tank per ANSOP 68_____ and insert spool piece under this tank. Close and lock WD-V-18.

6. Establish continuous monitoring of fuel element temperatures per ANSOP 5705.

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7. Check that cooling water is on the steam condenser EP 274.

8. If at any time up through step 23, the temperature difference between the fuel element and the flooding water (as measured by TI-203) exceeds 200°F, admit steam to the flooding water heater, EP 271.

If the above differential still exceeds 200°F, initiate steam cooling by:

a. Closing the following valves:

WP - V - 19
WP - V - 21
WP - V - 22
WD - V - 21
GNL- V - 14

b. Opening S-MV-201 and S-MV-110

Continue steam cooling as required to maintain the 200°F differential described above.

9. Shutdown pool water circulation per ANSOP 56_____.

10. Divert the main loop flow through the reactor bypass and close the main loop block valves by operating GNL-V-54 (open) and GNL-V-1 and GNL-V-2 (close).

11. Depressurize the reactor portion of the main gas loop per ANSOP 51_____.

12. Establish shutdown cooling by the shutdown cooling compressor per ANSOP 5717.

Note: If shutdown cooling is being provided by this means, steps 10 and 11 above shall be omitted.

13. Line up valves as follows:

CS-V-7	Closed
WD-V-19	Open
WD-V-20	Closed
WD-V-21	Open
WD-V-19	Closed-locked
WP-V-22	Open
WP-V-21	Closed-locked
GNL-V-49	Closed-locked
WP-MV-203	Closed-key in lock.

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14. Cycle WP-MV-203 to see that it operates properly.

15. From this point until the end of the procedure, the operator maintaining surveillance over the LCR indicators on Channel I and II shall immediately (a) execute ABLE, (b) discontinue flooding, (c) open valve WP-MV-203 to dump the water out of the reactor and (d) close S-MV-110 if it is open, anytime the LCR increases by a factor of 1.5 in one minute or less. In addition, if the LCR shows an increase of six times the base reading the above action shall be taken.

16. Establish base count rate on LCR. Record this value.

17. Hang "Danger" tags on the following rod control switches: SD-1, SD-2, Gang II, SB, S-1, S-2 and F.

18. Turn on the main console power and make up interlock and safety circuits, by-passing as required, to obtain magnet current. Do not bypass power or period circuits.

19. Set the rod selector switch on Safety Rod #1 and withdraw the safety rod. During withdrawal, observe the LCR, and, if the rate doubles, ABLE. Repeat the procedure with smaller increments of withdrawal. After #1 Safety Rod is out, repeat the procedure to withdraw #2 Safety Rod.

20. Discontinue shutdown cooling and close GNL-MV-103 and GNL-MV-104. Open GNL-V-14.

21. Open valve CS-V-10 and admit water to the bottom plenum. Observe the filling rate on FI-205. Observe level rise on LI-102.

22. Close WD-V-21 and lock it. Remove the spool piece under EP-252. Note: Do not confuse with WP-V-21.

23. Establish and record the new base count rate on the LCR system by three successive one minute readings.

Note: From this point forward, steam may not be introduced into the reactor. If the fuel elements overheat (more than 200°F above flooding water temperature), dump the reactor, reestablish cooling and repeat the procedure.

24. Hang "Danger" tags on the operating controls for S-MV-201 and S-MV-110.

25. Open WP-V-19. Adjust the rate with WP-V-20 to hold flow at 1 gpm or less. Confirm on LI-102 as a rate of rise of 0.5 in/min for 10 minutes and then a rise of 2.5 in/min for 11 minutes. If LCR readings are satisfactory it should require approximately 21 minutes to complete this step.

If LCR readings are not satisfactory as indicated in step 15, adjust fill rate by throttling WP-V-20.

26. When LI-102 indicates 8 ft. 2 in. total water level in the reactor, close WP-V-19 and WP-V-22. Lock both valves. Note: Do not confuse with WD-V-19.

27. Check that the LCR has been constant for the last 4 inches of filling.

28. Discontinue fuel element temperature surveillance and flooding water heating.

29. Open GNL-V-49 and admit water to upper plenum and piping.

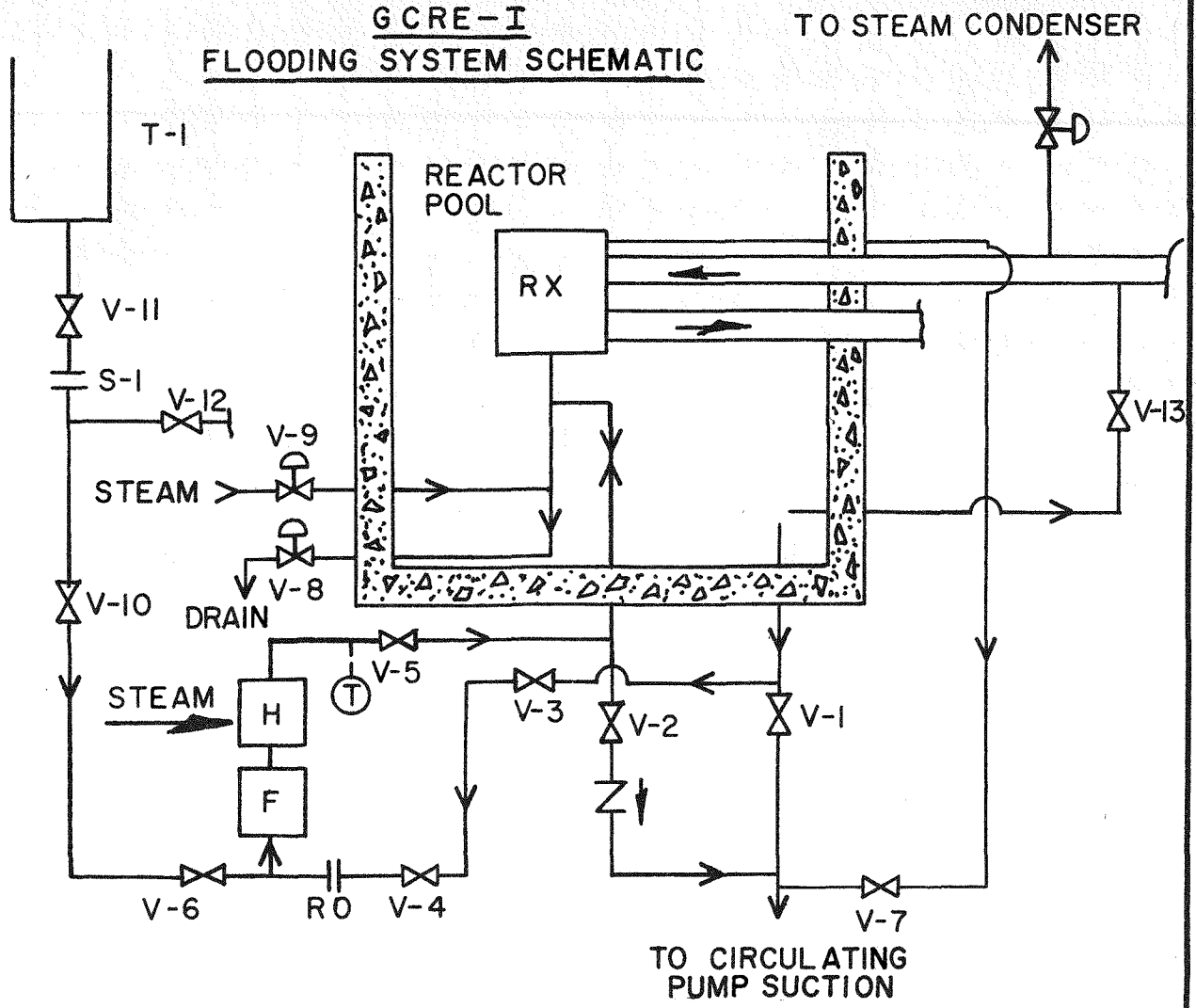
30. Reinsert the safety rods and turn off console power.

31. Discontinue LCR surveillance except for normal shutdown practice as described in ANSOP 5716.

32. Reestablish pool water circulation per ANSOP 56_____.

33. Proceed with reactor cover removal per ANSOP 5725.

GCRE-I
FLOODING SYSTEM SCHEMATIC



Eqpt.	Description	Normal Operation	Flooding Phase I	Flooding Phase II	Flooding Phase III
V-1	Manual Valve	Open	Closed	Closed	Closed
V-2	Manual Valve	Closed (L)	Closed (L)	Closed (L)	Closed (L)
V-3	Manual Valve	Closed (L)	Closed (L)	Open	Closed (L)
V-4	Need Valve set for 1 gpm	Set (L)	Set (L)	Set (L)	Set (L)
V-5	Manual Valve	Closed (L)	Open	Open	Closed (L)
V-6	Manual Valve	Closed (L)	Open	Closed (L)	Closed (L)
V-7	Remote op.vlv	Open	Closed	Closed	Closed
V-8	Remote op.vlv	Closed	Closed	Closed	Closed
V-9	Manual Valve	Closed	Closed	Closed	Closed
V-10	Manual Valve	Closed	Open	Closed	Closed
V-11	Manual Valve	Closed	Open	Closed	Closed
V-12	Manual Valve	Closed	Closed	Closed	Closed
V-13	Manual Valve	Closed (L)	Closed (L)	Closed (L)	Open
F	Flow Meter	Out of Service	In Service	In Service	Out of Service
1+	Water Heater	Out of Service	In Service	In Service	Out of Service
T-1	Cal.Fldg.Tank	Out of Service	In Service	Out of Service	Out of Service
S-1	Spool Piece	Removed	Installed	Removed	Removed
RO	Flow Rest.Orf	Out of Service	Out of Service	In Service	Out of Service
T	Temp.Indctr.	Out of Service	In Service	In Service	Out of Service

(L) - Denotes "Locked"

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STANDARD OPERATING PROCEDURE

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION:

TITLE: FUEL ELEMENT HANDLING

SCOPE

This procedure outlines the steps to be followed in normal fuel handling. The situations covered are:

A. Preparations for charging new fuel.
B. Preparations for charging partially depleted radioactive fuel.
C. Removal of radioactive fuel from the reactor to the fuel element storage rack.
D. Charging of fresh fuel with no increase in core size.
E. Replacement of instrumented fuel element.
F. Relocation of instrumented fuel element.

For the procedure for removal of a ruptured fuel element, see ANSOP 5724. The shipment of spent fuel elements is outlined in ANSOP . The addition of fuel to the core when the core size is increased must be performed under a special procedure.

Basically fuel element handling involves the manual manipulation of the elements using the special tools provided. For reasons of safety and accountability, accurate accounting is of paramount importance in this procedure. Fuel elements are identified by serial number and the whereabouts of each element is known, along with its history, at any time.

PROCEDURE

A. Preparation for Charging New Fuel

1. The Operating Superintendent will perform the following steps prior to the scheduled charging.

a. Discuss with the Assistant Project Manager, the scope of work to be accomplished in the fuel element handling comtemplated.

b. Prepare a draft of ANSOL 5758 completing columns 1, 3, 4 and 5, and column 9 as required.

c. Proceed to the vault and select the fuel elements to be charged. Insert the serial number of the element opposite the proper step in

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Column 2. Check to see that the proper orifice is in place in the element as shown in Column 3.

d. Prepare the final copy of ANSOL 5758 and submit to the Assistant Project Manager for approval.

2. The Assistant Project Manager will review and approve the ANSOL 5758.

B. Preparations for Charging Used Fuel

1. The Operating Superintendent will perform steps A-1-a and A-1-b of Section A above.

2. The Operating Superintendent will consult the Fuel Element Storage Rack Log and locate the elements to be handled. The serial number of the element will be entered on ANSOL 5758 opposite the proper step. If an orifice change is required, it shall be clearly noted in column 9.

3. The Operating Superintendent will perform step A-1-d of Section A above.

4 The Assistant Project Manager will review and approve the ANSOL 5758.

C. Removal of Spent Fuel from Reactor to Fuel Element Storage Rack.

1. Check that the reactor is shutdown per ANSOP 5715.

2. Check that shutdown neutron monitoring is in operation in accordance with ANSOP 5716.

3. Visually check that all rods are in the reactor.

4. Check that ANSOP 5720, Reactor Flooding, has been completed.

5. Initiate the portion of ANSOP 5716 - Shutdown Neutron Monitoring described for use during fuel handling.

6. Obtain the previously prepared fuel element handling schedule, ANSOL 5758.

7. Obtain a properly authorized SWP to cover the work to be performed. (See ANSOP 2300).

8. Check that the pool water level is normal.

9. Distribute personnel as follows:

a) Shift Supervisor - Control Room
b) Control Room Operator - Control Room
c) Fuel Handler and Charger - Work Bridge

10. Withdraw the safety rods per ANSOP 5718, steps 16 through 19.

11. Obtain the holddown manipulator tool and the fuel element handling tool and hang them from the mobile work bridge by the hooks provided.

12. Move the work bridge into position over the reactor. See ANSOP 5011 covering the operation of this bridge.

13. Locate the fuel element to be removed. Refer to the fuel element handling schedule, ANSOL 5758, and the core diagram, ANSOL 5759.

NOTE: Elements shall be handled in the sequence established in the fuel element handling schedule. No deviation from this sequence is permitted without the approval of the Operating Superintendent.

14. Using the holddown manipulation tool, release the holddown to free the fuel element to be removed.

15. Engage the fuel element handling tool to the top of the fuel element and raise the fuel element clear of the pressure tube and top plenum. Observe the requirements of the SWP during this operation.

16. Move the work bridge to position over the fuel element storage rack.

NOTE: An operator other than the one handling the fuel element must operate the bridge.

17. Locate the position in the fuel element storage rack for deposit of the fuel element using the fuel element handling schedule and the fuel element storage rack diagram, ANSOL 5759.

18. Lower the fuel element into the proper position in the storage rack and disengage the handling tool.

NOTE: If fuel is to be charged into the core at this point, execute steps D-1 through D-7 of section B of this ANSOP. If not, proceed with Step 19 below.

19. Return the work bridge to a position over the reactor and proceed with the removal of the next fuel element as indicated on the handling schedule. Repeat steps 13 through 18 until the handling specified on the schedule is completed.

20. When all handling is complete, proceed with step D-8 below.

D. Charging of Fresh Fuel

1. Prior to the initiation of this procedure, perform the below steps:

a. Obtain the handling schedule and proceed to the fuel element storage vault

b. Clear the ADT system per ANSOP and enter the vault.

c. Locate the first fuel element specified on the handling schedule by serial number.

d. Transfer this fuel element to the fuel element transfer cart.

e. Move the loaded fuel element transfer cart to the work area.

2. Check the serial number and orifice size of the element to be charged and complete columns 6 and 7 of ANSOL 5758.

3. Transfer the element to be charged to the work bridge.

4. Attach the fuel element charging tool to the fuel element and lower it into the pool.

5. Locate the core position into which the fuel element is to be charged as specified on the handling schedule. Check that the lock is clear.

6. Lower the element into the core position and disengage the tool.

7. Using the lock manipulation tool, lock the element into place.

NOTE: If the handling schedule specifies removal of a spent fuel element next, proceed with steps C-13 through C-18 of this ANSOP. If not, repeat steps D-2 through D-7 until all fuel elements are charged. Repeat step D-1 as required to obtain additional fuel elements from the vault.

8. Return the fuel element handling tools and the lock manipulation tool to their storage racks.

9. Position the bridge over the reactor and control rods.

10. Deliver the handling schedule to the Shift Supervisor. Terminate the SWP.

E. Replacement of Instrumented Fuel Element

1. Complete steps C-1 through C-12 above.

2. Obtain the following equipment, materials, and tools and assemble same in the work area or on the work bridge.

a. Instrumented fuel element with attached T/C harness. Leave the element in its protective container with the harness coiled.

b. Handling tools and pull lines.

c. Adapter and water seal with gasket.

d. Emerto collars.

e Marman Gasket #51303 - 22 - 23.

3. Position the work bridge over the reactor.

4. Bleed down the pressure on the T/C pressurized harness for the fuel element to be replaced.

5. Release the T/C pressurized harness to be replaced from the clamp at the pool edge.

6. Grappel and hook the lifting eye on the terminal box with the crane auxiliary hoist.

7. Drift the terminal box to the center of the pool and raise the terminal box to bridge level. Observe SWP requirements during this step. Secure the cable below the box to the bridge and remove the crane hook.

8. Break the Marman coupling on the terminal box and remove the cap screws at the top of the box. Discard the gasket.

9. Slide the top portion of the terminal box up out of the way.

10. Disconnect the T/C wires leading to the instrumented fuel element.

11. Break the Weatherhead coupling at the bottom of the terminal box and remove the harness from the terminal box. A hard pull will release the Emerto seal collar at this location. Retain the locking nut.

12. Install the adapter and water seal on the removed end and attach the pull line to the eye on the seal. Support the cable near its end from the crane hook.

13. Loosen the locking nut on the Emerto seal at the reactor flange using an extension and wrench.

14. Pull up sharply with swivel head tongs holding the harness just above the reactor flange to break the seal on the Emerto collar at this point.

15. Using the swivel head tongs, grasp the T/C harness inside the reactor and pull it through the pressure seal. Pay out on the crane hook to lower the sealed end as required.

16. When enough slack is available, release the locking dog on the fuel element to be removed, grasp the element with the fuel element handling tool, and remove it from the reactor. Leave it hanging in the handling tool from the bridge rail.

17. Continue step D-14 until all the cable is pulled through the pressure seal.

18. Grapple the pull line and bring it to the surface. Observe SWP precautions during this step.

19. Disconnect the pull line from the eye on the water seal and drop the T/C harness back into the pool.

20. Move the removed element to the designated storage position (see the handling schedule) and move the cable out of the way as much as possible.

21. Install an adapter and water seal on the new element harness.

22. Attach the end of the pull line which emerges from the reactor to the eye in the seal.

23. Attach the fuel element charging tool to the new element and lower it into the pool, outside and above the reactor. Secure the charging tool to the bridge rail.

24. Lower the pull line with the sealed end of the harness into the pool while pulling on the other end of the line. Support the weight with the crane using the special clamp.

25. Release the crane clamp and, using the swivel tongs and the pull line, feed the end of the harness up through the seal.

26. Reengage the crane hook on the harness above the pressure seal and continue to pull the harness through the seal.

27. Discontinue pulling and insert the fuel element in its proper location when enough of the cable is fed through the seal to make this step possible. Secure the fuel element lock and release and remove the fuel element charging tool.

28. Continue pulling the harness up until the proper amount of slack exists in the reactor top plenum. At this point the sealed end will be out of water (observe SWP conditions).

29. Secure the sealed end at the bridge and release the crane hook.

30. Make up the harness clamps inside the top plenum with the special tool provided for this purpose.

31. Remove the adapters and water seal from the exposed end of the harness.

32. Position the seal collar and nut at the pressure seal and make up about one turn.

33. With step D-30 complete, pull the slack through the pressure seal and tighten the seal nut with the wrench.

34. Slip an Emerto seal collar and the locking nut removed in step D-10 over the end of the harness and insert the harness into the bottom of the terminal box.

35. Make up the seal at the bottom of the box.

36. Make up the thermocouple wires to the terminal board.

37. Slip the top half of the terminal box into place and make up the Marman clamp and the cap screws. Use care to protect the upper seal gasket; insert a new lower seal gasket.

38. Pressurize the harness at the FPM panel and check for leaks. Use soap on the above water portion of harness.

39. Engage the crane hook in the terminal box lifting eye and release the securing lashing on the harness.

40. Lower the harness into the pool with the crane. Drift the terminal box into position and make up the pool edge clamp.

41. Disengage the crane hook.

42. Complete steps D-8 and 9 and 10 above.

F. Relocation of Instrumented Fuel Element

1. Follow steps E-1, E-2, E-3, E-4, E-5 and E-6 above.

2. Complete steps E-13, E-14 and E-15 above.

3. Release the locking dog on the element to be relocated and the harness clamps in the top plenum.

4. Grasp the element with the fuel handling tool, raise it, transfer it to the new position, insert it and release the tool.

5. Secure the fuel element lock and the harness clamps in the top plenum.

6. Complete steps E-33, E-37, E-38, E-39 and E-40 above.

7. Complete steps D-8, 9 and 10.

STANDARD OPERATING PROCEDURES

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: INTERCHANGE OF FINE & SHIM RODS

SCOPE

This procedure outlines the steps to be followed in making an interchange of the fine rod and a shim rod. This action will be necessary from time to time as the fine rod moves to extremes of its position as a result of xenon build-in or decay, changes in power level, etc.

In general, the procedure specifies the insertion of rod to place the reactor on a negative period followed by removal of the other rod to retain criticality.

The procedure is written as though the fine rod were all the way in and it is desired to withdraw it and replace it with the equivalent worth of shim rod #1. Obviously, the replacement may be made with shim rod #2 or some combination of the two shim rods by proper selection with the rod selector switch. Not so obviously, the interchange may be made in the reverse direction (to insert fine rod and withdraw shim rod) by reversing the order of rod callout in the procedure.

PROCEDURE

Note: This procedure shall not be undertaken unless the reactor has been at steady state conditions for at least five minutes.

1. Transfer the control of the fine rod to manual.

2. Select shim rod #1 on the rod selector switch.

3. During the interchange, reliance shall not be placed on a single power indicator since shadowing may distort the indication. All nuclear power indicators as well as the BTU indicator shall be used as an index of the power behavior of the reactor.

If at any time during the interchange, the reactor behaves in an unusual fashion, immediately discontinue the interchange and bring the reactor under control with the fine rod. Adjust power as necessary, wait the necessary five minutes and continue with the interchange.

The rod interchange must be made slowly and deliberately, with full knowledge of each next step and the predicted result of the step. The interchange will not be made under emergency conditions or conditions of high duress.

4. With one hand on the shim rod toggle switch and the other on the fine rod toggle switch, insert about one inch of shim rod.

5. Jog the fine rod control to remove sufficient fine rod to bring power back to nominal level.

Caution: The fine rod travels about 25 times as fast as the shim rods. The shim rods are worth about 2.5 times the fine rod in reactivity. All other things being equal a one sec. movement of shim rod would be compensated for by 0.1 sec movement of fine rod. However, both rods have about 50% of their worth in the first 20% of travel (from full in outward) so the relative position of the rods will affect the ratio. In summary, jog the fine rod and watch the power instruments.

6. Repeat steps 3 and 4 until the fine rod is withdrawn to the desired position.

7. Turn the rod selector switch to "off".

8. Adjust power and place the fine rod on servo control per ANSOP 5719.

STANDARD OPERATING PROCEDURES

GAS-COOLED REACTOR EXPERIMENT-I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: EMERGENCY SHUTDOWN PROCEDURES

SCOPE

This procedure outlines the two basic emergency shutdown procedures for the GCRE-I reactor. These are identified as ABLE and BAKER in order of decreasing shock to system and are referred to by these names throughout the normal and emergency operating procedures.

PROCEDURE

Able

1. Press the manual scram button on the center strip of the console. This will automatically drive in the six shutdown and two safety rods.

2. Observe that the neutron flux is decreasing on the Log-N, Linear, and Safety Channel indicators on the console.

3. Trip the setback rod, run in the fine rod and run in the shim rods in gang.

Baker

1. Press the manual setback button on the center strip of the console. This will automatically drive in the setback rod.

2. Observe that the neutron flux is decreasing.

3. If the flux is not decreasing, execute ABLE.

4. If the flux is decreasing, run in the shutdown rods and safety rods in two gangs.

5. If at any time the flux increases, execute ABLE.

6. Run the fine rod in.

7. Run the shim rods in ganged.

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STANDARD OPERATING PROCEDURES

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION EMERGENCY

TITLE: SCRAM, SETBACK AND ANNUNCIATOR CIRCUIT SCHEDULE

SCOPE

This procedure lists the scram, setback and annunciator circuits along with the actuating device, set point, annunciator panel location, and a brief description of the function of the circuit:

SCRAM CIRCUITS

No.	Description	Ann. Panel	Actuator	Normal Set Point
1	Manual Scram	C-4	Manual button	Normally closed; depress to open
2	Power failure	C-1	KPF Relay	Normally closed; opens on power failure
3	Gas flow low	C-8	FR-101	85% nominal flow
4	Gas outlet temperature-high	C-3	TR-103	1260°F
5	Power level - high	C-7	Safety amplifier	115% nominal power
6	Seismometer	C-2	Seismic switch	Normally closed, opens on seismic disturbance
7	Reactor room doors	C-5	K-3C	Normally closed, opens when any door is opened
8	Run-safe switch	C-6	K-2C	Normally closed (on run) open on safe.
9	Log N-high	C-12	Log N recorder	115% nominal power
10	Period-Short-Channels I, II, III or IV	C-11	Period recorder	7 seconds
11	Period-Short-Channels III and IV	C-10	Period amplifier	7 seconds
12	Safety rods, low pressure	D-13	PT-109	50 psig

SETBACK CIRCUITS

1	Period-short-Channels I and II	C-13	Period chassis, Channels I and II	10 seconds
2	Gas outlet temperature-high	C-14	TR-103	1240°F
3	Gas flow - low	C-16	FR-101	90%
4	Power level - high	C-15	Power level recorder	110% nominal power
5	Manual setback	C-9	Manual button	Normally closed, depress to open

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ANNUNCIATOR CIRCUITS

No.	Description	Ann. Panel	Actuator	Normal Set Point
1	Burner #1 - Flame failure	A-1	BS-1	Normally open, closes on burner failure
2	Burner #2 - Flame failure	A-2	BS-2	Normally open, closes on burner failure
3	Burner #3 - Flame failure	A-3	BS-3	Normally open, closes on burner failure
4	Fuel oil supply-low pressure	A-4	PS-311	105 psig
5	Fuel oil storage tank-low level	A-8	LT-306	5' level
6	Utility Water storage-low level	A-5	PS-306	10' level
7	Utility Water system-low press.	A-6	PS-302	80 psig
8	Utility water storage-high level	A-9	PS-306	19' level
9	Utility air system-low press.	A-7	PS-305	90 psig
10	Safety rods - low pressure	A-10	PI-401	100 psig
11	Steam boiler-flame failure	A-11	BASB	Normally open, closes on burner failure
12	Boiler fuel tank-high level	A-12	LS-350	90% capacity
13	S-201 outlet-high conductivity	A-13	CA-201	1 umho
14	S-301 outlet-high conductivity	A-14	CA-301	1 umho
15	Cooling tower basin-low level	A-15	LS-302	75% capacity
16	Cooling tower basin-high level	A-16	LS-302	95% capacity
17	Emergency power supply-failure	B-1	--	Normally open, closes on emerg. pwr. failure
18	Shutdown cooling loop-low flow	B-2	FT-107	7.5 cu. ft/min
19	Shutdown cooling loop-high temp	B-3	TI-111	400°F
20	Blowdown N_2-low pressure	B-4	PS-104	1500 psig
21	Fuel element-high temperature	B-8	TI-401	1550°F
22	Compressor-low speed	B-5	TACH-102	97% of set point
23	Compressor or gear-exit oil; high temperature	B-6	TI-116 TI-117	165°F
24	Compressor-lube oil, low press.	B-7	PS-117	10 psig
25	Main compressor-seal water-low pressure	B-14	PS-115	250 psig
26	Main compressor-oil or water reservoir low level	B-15	LS-103 LS-104	14' level 20.5' level
27	Main loop makeup-high flow	B-9	FT-102	0.1 cu ft/min.
28	Main loop - high humidity	B-10	HI-101	0°F dewpoint
29	N_2 stg - low pressure	B-11	PS-103	350 psig
30	Main loop - O_2 content high	B-16	OI-101	0.5% O_2
31	O_2 stg - low pressure	B-12	PS-105	350 psig
32	Instrument air-low pressure	B-13	PS-308	60 psig
33	Main loop - flow low	D-1	FR-101	95% of nominal
34	Main loop - high pressure	D-2	FR-102	110% of nominal
35	Main loop - low pressure	D-3	PR-102	90% of nominal
36	Rx outlet-temp high	D-4	TR-103	1220°F
37	Demineralized water system-low flow	D-5	FT-104	850 gpm

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ANNUNCIATOR CIRCUITS

No.	Description	Ann. Panel	Actuator	Normal Set Point
38	Demineralized water system high temperature	D-6	TI-108	190°F
39	E-102 W C outlet-high temp.	D-7	TI-107	110°F
40	Cooling water system-low press.	D-8	PS-301	50 psig
41	Waste N_2 storage tank-high level	D-9	LT-206	75% capacity
42	Reactor pool - low level	D-10	K-99A	6' below normal
43	Reactor pool - high level	D-11	LS-202	6" above normal
44	Demin. water surge drum - low level	D-12	LS-101	24" level
45	D-203 waste tank - high level	D-14	LI-207	72" level
46	D-202 waste tank - high level	D-15	LI-203	72" level
47	Demin. water surge drum - low pressure	D-16	PG-123	28 psig
48	Gas-water HX-outlet gas temperature high	E-1	TR-105	150°F
49	RAMS - High activity	E-2	RAMS meter relays (10)	Manual set
50	Nuclear instrument-failure	E-3	CIC power supply 1, 2 or 3	Failure
51	Water in main loop	E-4	LI-104	Normally open, closes on detection of water
52	T/C Manifold-high or low press.	E-5	K-1 switch	175 psig 125 psig
53	Inlet duct-high temp.	E-6	Meter contacts	400°F
54	Exhaust duct-high temp.	E-7	Meter contacts	600°F
55	Water in rods	E-9	Electrode detectors (14)	Normally open, closes on detection of water
56	FPM High	E-8	MS in recorder	Manual Set

DECLASSIFIED

STANDARD OPERATING PROCEDURE

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - SAFETY RODS - PRESSURE LOW

SCOPE

This procedure outlines the emergency action, to be taken when the annunciator sounds indicating a low pressure on the nitrogen supply to the safety rods drive cylinders. This annunciator is actuated by a mercoid switch in P1-401 and indicates the safety rod pressure is less than 100 psig.

PROCEDURE

Immediate Action

1. Dispatch an operator to check the nitrogen supply header pressure (PG-106-113). If the header pressure is below 300 psig, valve in a new bank of cylinders.

2. Check the reactor pool for gas bubbles indicating a leak in the underwater portion of the system. If gas bubbles are encountered, execute BAKER.

3. If the nitrogen supply header pressure is normal and no bubbles are indicated, check the mercoid switch P1-401 for malfunction.

4. If the condition cannot be corrected by the time the pressure reaches 75 psig, execute BAKER to avoid the scram which will occur automatically at 50 psig.

Subsequent Action

1. When the reactor is down, check the components on the nitrogen supply system as follows:

a. Check mercoid switch P1-401 for malfunction.
b. Check the pressure regulating valve for malfunction.
c Check nitrogen supply lines to the safety rods for indications of leakage.

STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 - REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - EMERGENCY POWER SUPPLY FAILURE

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating an emergency power supply failure. This annunciator is activated by relay and indicates a power failure in the emergency supply bus.

PROCEDURE

Immediate Action

1. Check the following equipment for possible stoppage due to the loss of the emergency diesel generator.

a. If the main compressor (C-101) is off, execute ABLE.

b. If both of the demineralized water pumps (P-101 A and B) are off and the standby pump fails to start manually, execute ABLE.

c. If both of the cooling water pumps (Pg-304 A and B) are off and the standby pump fails to start manually, execute ABLE.

d. If the seal water pump (C-101) is off and auxiliary seal water pump fails to start manually, execute BAKER.

e. If the lube oil pump (C-101) is off and auxiliary lube oil pump fails to start manually, execute BAKER.

2. If the above equipment continues to operate proceed to the diesel generator unit and check the position of switch TS-1 to verify the annunciator. If the diesel generator unit is off the line proceed as follows:

a. Check the diesel unit for possible mechanical or electrical failures. If any are encountered, execute BAKER.

b. Check the diesel oil supply tank; if low, refull.

c. Check the diesel generator speed and frequency. If the speed and frequency are correct, switch TS-1 from normal to emergency bus. If the diesel generator fails to maintain the load, execute BAKER. If the

diesel generator unit maintains the load, repeat step 1.

Subsequent Action:

1. Check the switch TS-1 for malfunction

2. Check the diesel generator unit for mechanical and electrical malfunction.

STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - BLOWDOWN N_2 SYSTEM - LOW PRESSURE

SCOPE

This procedure outlines the emergency action to be taken in case the annunciator sounds indicating low pressure on nitrogen blowdown system. This annunciator is actuated by PS-104 located at nitrogen blowdown storage banks.

PROCEDURE

Immediate Action

1. Dispatch an operator to check the nitrogen pressure on the nitrogen blowdown storage bank. If header pressure is below 1000 psig, valve in new bank of nitrogen.

2. If the above does not clear the annunciator or if the annunciator cannot be by other means within 5 minutes from original annunciator, execute BAKER.

Subsequent Action

1. Check components of nitrogen blowdown storage system as follows:

   a. Check PS-104 for malfunction

   b. Check GNL-PSV-101 for malfunction

   c. Check nitrogen blowdown storage system where possible for indication of leakage.

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STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT-I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - FUEL ELEMENT - HIGH TEMPERATURE

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a fuel element high temperature. This annunciator is actuated by TI-401, indicating a fuel element temperature greater than 1550°F.

PROCEDURE

Immediate Action

1. Check the fission product monitor for indication of high activity. If the fission product monitor is increasing rapidly, execute ABLE. If it is normal or increasing slowly proceed to step 2.

2. Within 5 minutes, check the following:

a. Check reactor power on BTUI-101.

(1) If the power is normal, check TI-103 for evidence of malfunction of the power indicating system. If TI-103 is normal, proceed to step b; if TI-103 indicates high power, reduce power level until TI-103 is normal.

(2) If BTUI-101 indicates that the power is high, decrease to normal operating level.

b. Check FR-101 for low gas flow. If flow is normal proceed to step c. If flow is low, increase speed of C-101. If flow cannot be increased, execute ABLE.

c. Check TIC-104 for high outlet temperature from FH-101. If the outlet temperature is high, decrease control point of TIC-104.

3. If the above does not clear the annunciator or if the annunciator cannot be cleared by other means within 5 minutes or if fuel element temperature reaches 1600°F, execute ABLE.

Subsequent Action

1. Contact Day Supervision.

2. After the shutdown is complete, resolve the difficulty and take the necessary corrective action.

STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - MAIN COMPRESSOR - LOW SPEED

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a low speed on the main compressor unit. This annunciator is actuated by Tach 102, and indicates main compressor speed is less than 95% of set-point.

PROCEDURE

Immediate Action

1. Check the speed of main compressor unit at Tach 102 on the console to verify annunciator. If the Tach 102 indicates a low speed, increase the set point till the main compressor speed is at normal speed.

2. If the above action does not clear the annunciator, execute BAKER.

Subsequent Action

1. Check the components of main compressor unit as follows:

   a. Check the main compressor unit for mechanical or electrical failure.

   b. Check the magnetic coupling for malfunction.

   c. Check the Tach 102 and set point for malfunctions.

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STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - COMPRESSOR OR GEAR EXIT OIL HIGH TEMPERATURE

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a high temperature in the compressor or gear exit lube oil. This annunciator is actuated by TI-116 or TI-117 and indicates an exit temperature greater than 165°F.

PROCEDURE

Immediate Action

1. Dispatch an operator to check TI-303 and TI-304 in the compressor and gear bearings lube oil exit lines. If the temperatures are normal, check TI-116 and TI-117 for malfunction.

2. If one temperature is high and one temperature is normal, contact the control room on the intercom and report the condition and request an ABLE. Execute step 6 below.

3. If both exit oil temperatures are high, check lube oil cooler for malfunction. Look for leaks, evidence of line pluggage. If all appears normal, check TI-303 and TI-304 again. If the temperature has risen to 190°F, contact the control room on the intercom and request an ABLE and execute step 6 below. If the temperature is less than 190°F, proceed immediately with step 4.

4. Check cooling water supply header pressure at PG-307 and PG-308. If supply header pressure is low, request the control room to switch to the standby pump.

5. If the above action does not clear the annunciator or if the annunciator cannot be cleared by other means within 15 minutes, execute ABLE and proceed to step 6 below.

6. Shutdown the compressor and initiate blowdown per ANSOP -5717

Subsequent Action

1. Contact Day Supervision

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2. Check components of lube oil system as follows:

a. If the cooling water system was at low pressure, determine why and check interlock (PS-301) and second pump starting circuit.

b. If the cooling water system pressure was correct, check WC-TCV-302 for malfunction.

c. If the above does not reveal the source of trouble, rod or snake the cooler water tubes for pluggage.

d. Inspect the compressor and/or gear unit bearings if the exit oil temperature reached 220°F. Scrape or repour as required.

STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - MAIN LOOP - OXYGEN CONTENT HIGH

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a high oxygen content in the main loop. This annunciator is actuated by OI-101 and indicates the oxygen content in the main loop is greater than 0.5%.

PROCEDURE

Immediate Action

1. Check the OI-101 indicator in the control room to verify annunciator. If OI-101 indicates oxygen content in the main loop is greater than 0.6%, execute BAKER. If OI-101 indicates oxygen more than 0.5% and less than 0.6% proceed as follows:

   a. Decrease OIC-101 set point to lesser position.

   b. Decrease the set point of PIC-102 to about 8% below the normal operating level, increase the set point of FIC-108 till the main loop pressure is at normal operating level. This action will allow the main loop to purge through GNL-PCV-102C. Continue this action until the oxygen content in the main loop is below 0.5%. When the oxygen content is normal, decrease the setpoint of FIC-108 to zero and increase the setpoint of PIC-102 to the normal operating level.

2. If the above action does not clear the annunciator within 15 minutes or if the oxygen content exceeds 0.6%, execute BAKER.

Subsequent Action

1. When the reactor is down, check the components of the oxygen and humidity analyzer and GO system as follows:

   a. Check oxygen and humidity analyzer for malfunction.

   b. Check OI-101 for malfunction.

   c. Check GO-FCV-101 for malfunction.

STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - DEMINERALIZED WATER SYSTEM HIGH TEMPERATURE

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a high temperature in the demineralized water system. This annunciator is actuated by TI-108 and indicates a demineralized water discharge temperature from E-101 greater than 190°F.

PROCEDURE

Immediate Action

1. Check TIC-105 set point, (E-101 gas bypass). Reduce the set point to zero bypass flow. If this does not clear the annunciator proceed to step 2. If TI-108 reaches 205°F, execute BAKER.

2. Check FI-109 (Demineralized water flow rate).

a. If FI-104 is normal, check TI-107 (Cooling water outlet temperature from E-101).

(1) If TI-107 is normal, proceed to step 3.

(2) If TI-107 indicates a high temperature, check the cooling water supply header pressure at PC-307 and PC-308. If the supply header pressure is normal proceed to step 3. If the pressure is low, request the control room to switch to the standby pump. If this does not clear the annunciator proceed to step 3. If TI-108 reaches 205°F, execute BAKER.

b. If FI-104 is low, switch to the standby pump (101A or B). If the flow rate remains low, check the nitrogen pressure in D-101. If the pressure is low, increase to normal. If this does not clear the annunciator proceed to step 3. If TI-108 reaches 205°F, execute BAKER.

3. If the above action does not clear the annunciator or if the annunciator cannot be cleared by other means within 15 minutes or if outlet temperature exceeds 205°F, execute BAKER.

Subsequent Action

1. Contact Day Supervision.

2. When the reactor is down, check the components of demineralized water system as follows:

a. Check TI-108 for malfunction.

b. Check FI-104 for malfunction.

c. Check TIC-105 for malfunction.

d. Check P-101 A & B for mechanical or electrical failures.

e. Check WD system for indication of leakage.

f. Check the N_2 pressurizing system on D-101.

3. Isolate the trouble and repair as required.

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STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - REACTOR POOL - LOW LEVEL

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a low level in the reactor pool. This annunciator is actuated by LS-202 and indicates the pool level is 6 inches below normal.

PROCEDURE

Immediate Action

1. Dispatch an operator to verify the reactor pool level. If the reactor pool is low, proceed as follows:

   a. If the reactor pool level is dropping rapidly, execute BAKER.

   b. If the radiation level at the reactor pool edge reaches 500 mr/hr, execute BAKER.

2. If the reactor pool level is low and dropping slowly, check WD-LCV-202, P-202, the demineralized water system, and the reactor pool for leaks.

3. If the reactor pool level is normal, check LS-202 for malfunction.

   a. If the reactor pool level continues to drop below 1 foot, or if the annunciator cannot be cleared within 15 minutes, execute BAKER.

Subsequent Action

1. When the reactor is down, check the components of the reactor pool as follows:

   a. Check WD-LCF-202 for malfunction.

   b. Check LS-202 for malfunction.

   c. Check LC-202 for malfunction.

   d. Check the service lines connected to the reactor and reactor pool for indication of leakage.

STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT- I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - DEMINERALIZED WATER SURGE DRUM LOW LEVEL

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a low level in the demineralized water surge drum. This annunciator is actuated by LS-101 and indicates a 24" heel or less.

PROCEDURE

Immediate Action

1. Dispatch an operator to check the water level in the demineralized water surge drum. If the water level is at the normal level or above check LS-101 for malfunction.

2. If the water level in demineralized water surge drum is below normal operating level, execute BAKER.

Subsequent Action

1. When the reactor is down - proceed as follows:

   a. Shutdown demineralized water pumps P-101 A and B.

   b. Depressurize demineralized water surge drum by closing GN-V-7 and opening GN-V-11 slowly.

   c. Repair any malfunction encountered within the demineralized water system.

   d. Open WD-V-16 and WD-V-13 in line WD-5 and proceed to refill demineralized water surge drum to normal operating level.

   e. When the proper water level has been established in demineralized water surge drum, close GN-V-11 and open GN-V-7 slowly.

   f. When demineralized water surge drum pressure is established at 35 psig, start demineralized water pump P-101A or B and circulate demineralized water through the system.

   g. If the water level remains constant in the demineralized surge drum after 10 minutes of circulation, the system is ready for operation.

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STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - RAMS - HIGH ACTIVITY

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a high activity in ram system. This annunciator can be actuated by one or a series of radiation monitors.

PROCEDURE

Immediate Action

1. Check the ram system rack 17, panels 4 and 5, to locate the radiation monitor that is high and to determine the radiation level.

2. If the radiation level is increasing at a rate greater than 5/R in 15 minutes, or if the level exceeds 50/R/hr, execute BAKER.

Subsequent Action

1. When the reactor is down:

   a. Survey the area adjacent to the radiation monitor which caused the annunciator to verify the reading.

   b. If verified, consult the supervision concerning corrective action.

   c. If not verified, correct the malfunction in the RAM system.

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STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - NUCLEAR INSTRUMENT FAILURE

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a nuclear instrument failure. This annunciator is actuated by CIC power supply 1, 2 or 3 and indicates a power failure.

PROCEDURE

Immediate Action

1. Check the positions of safety rods. If safety rods are in, proceed with ABLE.

2. Check the CIC power supplies in rack 8, panels 6 and 7 for normal output. If normal, continue operation and proceed to step 3. If abnormal, execute ABLE.

3. Check the CIC power supply in rack 9, panel 5. If normal, proceed with subsequent action. If abnormal, switch the fine rod to manual and continue operation.

Subsequent Action

1. Remove and repair the faulty CIC power supply.

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STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - WATER IN MAIN LOOP

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating water in the main loop. This annunciator is actuated by L1-104, which indicates water is present in WP-14.

PROCEDURE

Immediate Action

1. Check the main loop gas flow rate at FI-101. If the loop flow is below normal rate, and the compressor speed is normal, execute ABLE.

2. Check the reactor power level and control rod movement. If the power level is up and control rod is driving in, execute ABLE.

3. Check the reactor pool for low level and bubbles. If either are present execute ABLE.

4. Open WP-MV-203 for 5 seconds and then close. If the annunciator clears, execute BAKER . If the annunciator does not clear, execute ABLE.

Subsequent Action

1. Contact Day Supervision.

2. When the reactor is down, enter the locked area and check the components in the reactor pool and adjacent lines as follows:

a. Valve in LI-102 and confirm presence of water in the system. If confirmed, initiate a blowdown. Close the main loop block valves, depressurize the loop and establish shutdown cooling as soon as possible. Control the water level in the system below 18" on LI-102 by opening WP-MV-203 as required to blow out water.

b. Check LI-104 for malfunction.

c. Check the following valves:

WP-V-21 Closed - locked	WP-V-14 Closed - locked
WP-V-22 Closed - locked	WD-V-21 Closed - locked
GNL-V-49 Closed - locked	

d. Isolate the trouble and repair as required.

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ANSOP 5874
Published 12-22-58

STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - THERMOCOUPLE MANIFOLD - PRESSURE - HIGH

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating a high pressure in thermocouple manifold system. This annunciator is actuated by K-1 pressure switch, indicating manifold pressure greater than 175 psig.

PROCEDURE

Immediate Action

1. Dispatch an operator to check the manifold pressure gauge. If the pressure is normal, check K-1 pressure switch for malfunction.

2. If manifold pressure is high, close the air inlet valve to the manifold system. If the pressure drops off, check the pressure control valve for malfunction.

3. If manifold pressure remains high with air inlet valve closed, this indicates that a pressure seal at the reactor is leaking. Execute BAKER.

Subsequent Action

1. Contact Day Supervision.

2. Apply pressure to the manifold system.

3. Valve off one harness. Close the supply air valve. If the pressure holds, the valved off harness is the leaker. If the pressure drops, it is not the leaker.

4. Continue steps 3 and 4 until the leaking harness is located and confirmed.

5. If the leaking harness is T-TC-7, T-TC-8, T-TC-9 or T-TC-10, proceed to replace the instrumented fuel element and harness involved.

6. If the leaking harness is B-TC-4, B-TC-5 or B-TC-6, consult Day Supervision.

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STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION: 5 REACTOR OPERATION

TITLE: EMERGENCY PROCEDURE - WATER IN THE RODS

SCOPE

This procedure outlines the emergency action to be taken when the annunciator sounds indicating water is in the rods. This annunciation is actuated by relay and indicates water in the rod housing or rod drain tank.

PROCEDURE

Immediate Action

1. Check rack 4, panel 7, to determine if the water is in the rod housing unit or in the drain tank.

2. If the indicator lights indicates water in the rod housing, execute ABLE.

3. If the indicator light indicates water in the drain tank proceed as follows:

   a. Operate AI-MV-307 and 308 to blow out the tank. Observe that the annunciator clears.

   b. Observe the time the drain tank takes in refilling to determine the magnitude of the leak.

   c. Discuss with day supervision the mode of action to be taken.

Subsequent Action

1. When the reactor is down determine the source of leakage and repair.

STANDARD OPERATING PROCEDURE

GAS-COOLED REACTOR EXPERIMENT - I

PROCEDURE SECTION 5: REACTOR OPERATION

TITLE: INITIAL CRITICAL EXPERIMENT

SCOPE

This procedure outlines the steps to be followed in the initial determination of the cold critical loading for the reactor. The procedure terminates with a measurement of the period to determine the excess reactivity of that core loading which is less than one full fuel element supercritical.

The initial critical loading will be determined in the wet condition. The critical assembly experiments have indicated that a symmetrical loading of approximately 37 elements will be required to attain wet criticality. After proper preparation, this procedure specifies incremental additions of reactivity (by adding fuel elements in specified locations) until criticality is attained. Inverse multiplication plots are maintained through the procedure to permit evaluation of the progress and to predict the final critical loading. These plots are compared with predicted plots of inverse multiplication to anticipate non-standard situations.

Extrapolation of the data developed in this experiment to the dry cold critical is considered to be quite straightforward. The initial dry critical experiment is outlined in ANSOP 9201.

PROCEDURE

1. Prerequisites for the initiation of this procedure are:

a. Construction complete and all construction contractor personnel excluded from the area.

b. Check-out of the facility and equipment complete.

c. Exclusion area established and guard force on duty.

d. Master Emergency Plan (ANSOP 3900) shall be in effect and personnel trained in its execution.

2. Personnel required for this test shall include at least the following:

Assistant Project Manager
Operating Superintendent
Health and Safety Officer
Nuclear Engineer
Two Shift Supervisors
Seven Technicians

3. A temporary start-up control center shall be established in the contaminated work area as follows:

a) The center shall be located so that personnel within its confines are out of the line of sight of the reactor. Provide necessary desks, tables, etc to accommodate the equipment and provide work space for plotting and calculations.

b) Provide a clock with sweep second hand.

c) Provide an interval timer (electric, capable of readings to nearest second).

d) Provide a calculating machine.

e) Provide an intercom on open circuit to control room.

The area shall be roped off to outline the physical limits.

4. Install aluminum scaffold in the reactor pit to form a working platform level with the top flange of the top plenum. This platform is to consist of a work space on the side of the reactor over the control rods. The work spaces are to be at least 3' wide and 6' long with the 3' edge adjacent to the reactor. Handrails are to be installed on all edges except the ones which adjoin the reactor. Access shall be provided from one work space to the other. A ladder shall be provided from the west work space to the pool edge.

5, Install temporary start-up counting circuitry as follows:

a) Insert water-tight aluminum thimbles into positions E-1, E-9 and A-3.

b) Install BF_3 detectors in the aluminum thimbles so that the chamber centerline is coincident with the nuclear midplane of the reactor.

c) Connect the detectors to three scalers located in the temporary control center.

6. Modify the permanent neutron circuitry as follows:

a) Connect a temporary scram button into the scram circuit and run it to the temporary control center. This involves running a single wire from terminal GG-228 to one contact of the temporary (normally closed) button and a return wire from the other button contact to terminal F-173 and removing the existing wire from GG-228 to F-173.

b) Set the scram trip points on the composite safety amplifiers as far downscale as the adjustment will permit.

c) Set the scram microswitch on the Log-N recorder at 0.1 on the scale.

7. Modify the rod interlock circuitry to permit operation as follows:

a) Gang II withdrawal and insertion with all other rods out.

b) Fine rod and setback rod operation with all other rods out.

8. Remove the top cover of the reactor per ANSOP 5720.

9. Fill the pool to within one inch of the lower side of the scaffold planks per ANSOP 56_____

10. Fill the reactor with demineralized water per ANSOP 5718 to within 2 inches of the top plenum top flange.

11. Insert the startup source in its holder into position E-5.

12. Start-up and check out the neutron detection system per ANSOP 5709 and source check the BF_3 systems.

13. Prepare the Fuel Element Handling Schedule (ANSOL 5758) and the fuel preparation portion of ANSOP 5721, Handling of Fuel Elements.

14. Complete the rod operability check per ANSOP 5710.

15. Complete the safety circuit check per ANSOP 5707.

16. Complete an SWP for the job per ANSOP 2200. Check that the SWP requires constant monitoring on the work scaffold.

17. Assign personnel as follows:

Function	Number	Location	Type
Fuel Element Chargers	2	Scaffold in Pool	Technicians
Control Room Operator	1	Control Room	Shift Super.
Control Center Super.	1	Control Center	Shift Super.
Scaler Operators	3	Control Center	Technicians
Plotter	1	Control Center	Technician
Fuel Element Handler	1	Vault-Pool Edge	Technician

NOTES: (1) From this point until the end of the procedure, each step must be authorized by the Control Center Supervisor by specific directive to the responsible individual. The Control Center Supervisor will complete ANSOL 5727 after each step.

(2) During this period, the Asst. Project Manager, the Operating Superintendent, the Health & Safety Officer and the Nuclear Engineer will observe the work and monitor the procedure. Any one of these is authorized to stop the work at any time. The Asst. Project Manager must authorize continuance after such stoppage. One of the above four must be in the Control Center at all times.

(3) Man limits in the critical areas are established as follows:

On the work scaffold	3 total
Control Center	Assigned crew plus 4 total

In addition, personnel in the control room or other areas shall conduct themselves at all times in such a fashion so as not to interfere with the conduct of this most important task.

(4) If the procedure from this point to criticality is interrupted for longer than four hours, repeat steps 12, 14 and 15 before proceeding.

18. Make up the interlock and safety circuits and energize rod magnets per applicable portions of ANSOP 5710. (Control Room Operator).

19. Withdraw Gang I to the full out position. (Control Room Operator).

20. Move ten fuel elements in their shipping containers to the pool edge. Using the auxiliary hoist on the crane, lower the shipping containers and fuel elements to the scaffold. (Fuel element handler). Note: In this connection, the handling of fuel elements shall be strictly in accord with the provisions of this procedure. This prohibits the presence of more fuel elements on the work platform than are scheduled for charging in the next immediate step.

21. Charge the ten fuel elements into positions E-4, D-5, D-6, E-6, F-6, F-5, F-4, D-4, D-7 and F-7 in that order. (Fuel element charges). (Elements 1-10).

22. Clear the reactor pool of all personnel and withdraw Gang II (Control Room Operator).

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23. Take three one-minute counts on the BF_3's with a one-minute wait between counts. (Scaler Operators).

Analyze these data to ascertain if the reactor is on a positive period. If a period is indicated, press the scram button. If not, proceed with the next step. (Control Center Supervisor).

24. Withdraw the setback rod. (Control Room Operator)

25. Repeat step 23.

26. Withdraw the shim rods ganged (Control room operator).

27. Repeat step 23.

28. Withdraw the fine rod. (Control room operator).

29. Establish the base count rate by taking a 10-minute (or 10,000 total counts, whichever occurs first) count with the BF_3 systems. (Scaler Operators).

30. Insert Gang II (Control room operator).

Note: During the following phase, fuel elements are added two at a time until a total of 18 is reached. If the 1/M plot reaches .2 before these 8 elements have been added, discontinue adding two elements at a time and follow the same procedure adding one element at a time.

31. Repeat step 20 with two fuel elements. (Fuel Element Handler).

32. Charge the two fuel elements into positions E-3 and E 7. (Fuel Element Changers) (Elements 11 and 12).

33. Clear the reactor pool of all personnel and withdraw Gang II. (Control room operator).

34. Measure the count rate by taking a 10 minute count as specified in step 29. Log on ANSOL 5726. (Scaler Operators).

35. Plot the inverse multiplication on the ANSOL 5725 which has been prepared for this experiment. Inverse multiplication (1/M) is equal to the old (in this case, base) count rate divided by the new count rate. Note that rates are used; total count must be divided by counting time in minutes to determine counts/minute (Plotter).

36. Insert Gang II (Control Room Operator).

37. Repeat step 20 with two fuel elements (Fuel Element Handler).

38. Charge the two fuel elements into positions C-5 and G-5 (Fuel Element Changers). (Elements 13 and 14).

39. Clear the reactor pool of all personnel and withdraw Gang II. (Control Room Operator).

40. Repeat steps 34 and 35.

41. Insert Gang II. (Control Room Operator).

42. Repeat steps 20 with two fuel elements (Fuel Element Handler).

43. Charge the two fuel elements into positions C-6 and G-4. (Fuel Element Changers) (Elements 15 and 16).

44. Repeat steps 33, 34, 35 and 36.

45. Repeat step 20 with two fuel elements. (Fuel Element Handler).

46. Charge the two fuel elements into positions C-4 and G-6. (Fuel Element Changers). (Elements 17 and 18).

47. Repeat steps 33, 34, 35 and 36.

NOTE: For the balance of the procedure, one fuel element will be added one at a time.

48. Call the Fuel Element Chargers out of the pool.

49. Remove the source from position E-5 and install it in position I-3.

50. Withdraw Gang II and repeat steps 34 and 35. Insert Gang II, setback rod and fine rod.

51. Repeat step 20 with one fuel element (Fuel Element Handler).

NOTE: The Control Center Supervisor must obtain approval from the Asst. Project Manager before proceeding beyond this point.

52. Insert the element in position E-5 (Fuel Element Chargers.) (Element 19)

53. Clear all personnel from the reactor pool and withdraw Gang II (Control room operator).

54. Repeat step 23.

55. Withdraw the setback rod.

56. Repeat step 23.

57. Withdraw the Fine Rod (Control room operator).

58. Repeat steps 34 and 35.

59. From this point until criticality is reached, the following check-in will occur after each fuel element is inserted and the count taken and plotted:

a) Plotter - call out the 1/M
b) Plotter - call out the number of elements charged
c) Plotter - call out the extrapolated critical number of fuel elements.
d) Control room operator - report readings on:
   (1) Channel I and II log count rate meters
   (2) Log-N recorder
e) Control Center Supervisor - log all the above data

60. Insert the setback rod and fine rod. (Control Room Operator)

61. Transfer one fuel element to the scaffold. (Fuel Element Handler)

62. Charge the fuel element into position C-3. (Fuel Element Chargers) (Element 20)

63. Clear all personnel from the pool and withdraw the setback rod.

64. Repeat step 23.

65. Withdraw the fine rod.

66. Repeat steps 34 and 35.

67. Repeat steps 60 through 66, charging fuel into positions in the following sequence until criticality is attained:

Step	Position	Element	Step	Position	Element	Step	Position	Element
a	G-7	21	j	B-4	30	s	G-2	39
b	C-7	22	k	H-5	31	t	C-2	40
c	G-3	23	l	B-5	32	u	G-8	41
d	D-3	24	m	H-4	33			
e	F-8	25	n	B-3	34			
f	F-3	26	o	H-6	35			
g	D-8	27	p	B-6	36			
h	E-2	28	q	H-3	37			
i	E-8	29	r	C-8	38			

CAUTION: Beginning with the loading of the element into B-5 or the attainment of a 1/M of 0.02 whichever occurs first, the following additional steps will be taken.

a. The Control Center Supervisor will obtain approval of the Asst. Project Manager before initiating the sequence of steps to accomplish the loading of each next element in series.

b. The fine rod will be withdrawn only 7 inches at which time step 23 will be executed. The rod will then be completely withdrawn and steps 58, 59 and 60 completed.

c. After loading and prior to rod withdrawal, the Asst. Project Manager shall direct the Control Center Supervisor to clear all personnel to the control room if analysis of the data indicates that criticality is possible with the then existing loading.

All personnel will be continually alert for the attainment of criticality as evidenced by one or more of the following:

a. Plotter - 1/M of essentially zero

b. Control room operator - sustained period on the period recorder or - sustained rise on the count rate recorder.

Any of the foregoing indications or any other unusual occurrence shall be reported to the Control Center Supervisor.

66. Once criticality is attained, insert the fine rod.

67. Withdraw the fine rod and measure the period as indicated on the period meter, period recorder, and log count rate meters. CAUTION: Do not allow the power to rise above 0.05 on the Log-N recorder.

68. Insert the fine rod and run all other rods in.

69. Establish shutdown neutron monitoring per ANSOP 5716 or execute ANSOP 5727 Overnight Lockup.

70. Remove the temporary instrumentation and return the altered systems to normal condition.

APPENDIX G

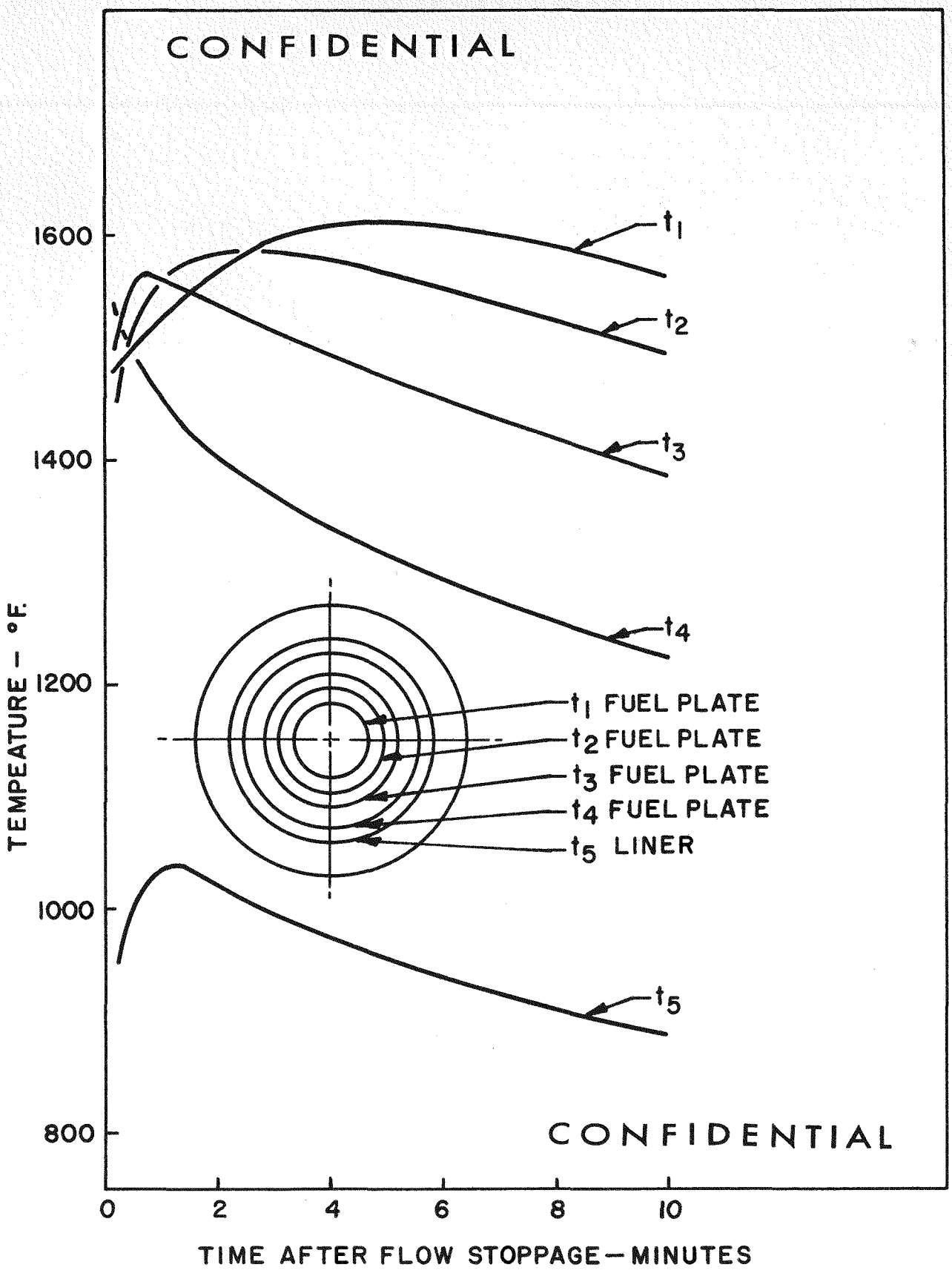
TRANSIENT GCRE I FUEL ELEMENT TEMPERATURES

To consider the effect of a coolant flow stoppage accident, an IBM 650 computer was used to calculate the surface temperature of individual fuel plates and insulation liners at a horizontal plane of maximum power generation. (X/L = 0.5). The central fuel element was selected for analysis, since it represents the hottest element in the radial array.

The fuel element model used in this analysis was the BMI Mark II element. All power plots (radial, axial, and intercell) were theoretical. The corresponding steady state plate temperatures for this element at the required axial position are as much as 100°F higher than for the GCRE element. The assumption was made that the reactor is scrammed one second after a coolant flow stoppage (which is quite conservative).

The Way-Wigner beta decay curve ($t^{-1.2}$) was utilized to calculate the beta energy level following a reactor scram. A more accurate beta decay curve (from an article by Thonton and Houghton in Journal of Applied Physics, November 1953), later indicated that the Way-Wigner analysis overestimated the fuel element heating by 10 to 50%. The heat generation rates in the individual fuel plates are shown in Figure G-2. The resulting fuel element temperatures are shown in Figure G-1. The calculations were made to cover the period including 10 minutes after reactor scram.

This analysis shows that the fuel elements will not melt down if the reactor is scrammed within approximately one second after a complete loss of coolant flow. This is because the heat transfer through the insulation into the water is sufficient to prevent excessive temperature rise from delayed heat generation in the fuel plates. The peak in the liner temperature results from the initial transient unbalance of radiant energy to the liner from the plates and conductive heat transfer out of the system through the insulation. This unbalance is corrected with time and the curve follows the usual shape. This occurs in the innermost concentric ring five minutes after shutdown. Because the analysis neglected connective cooling, axial conduction, and conduction out through the clips, it is believed that an estimate of 1610°F for the maximum temperature for this element is conservative. For the GCRE I element, due to the lower initial temperature and the various physical assumptions on the model, it is believed that the maximum temperature for this condition of flow stoppage should be about 100°F less than that shown above.



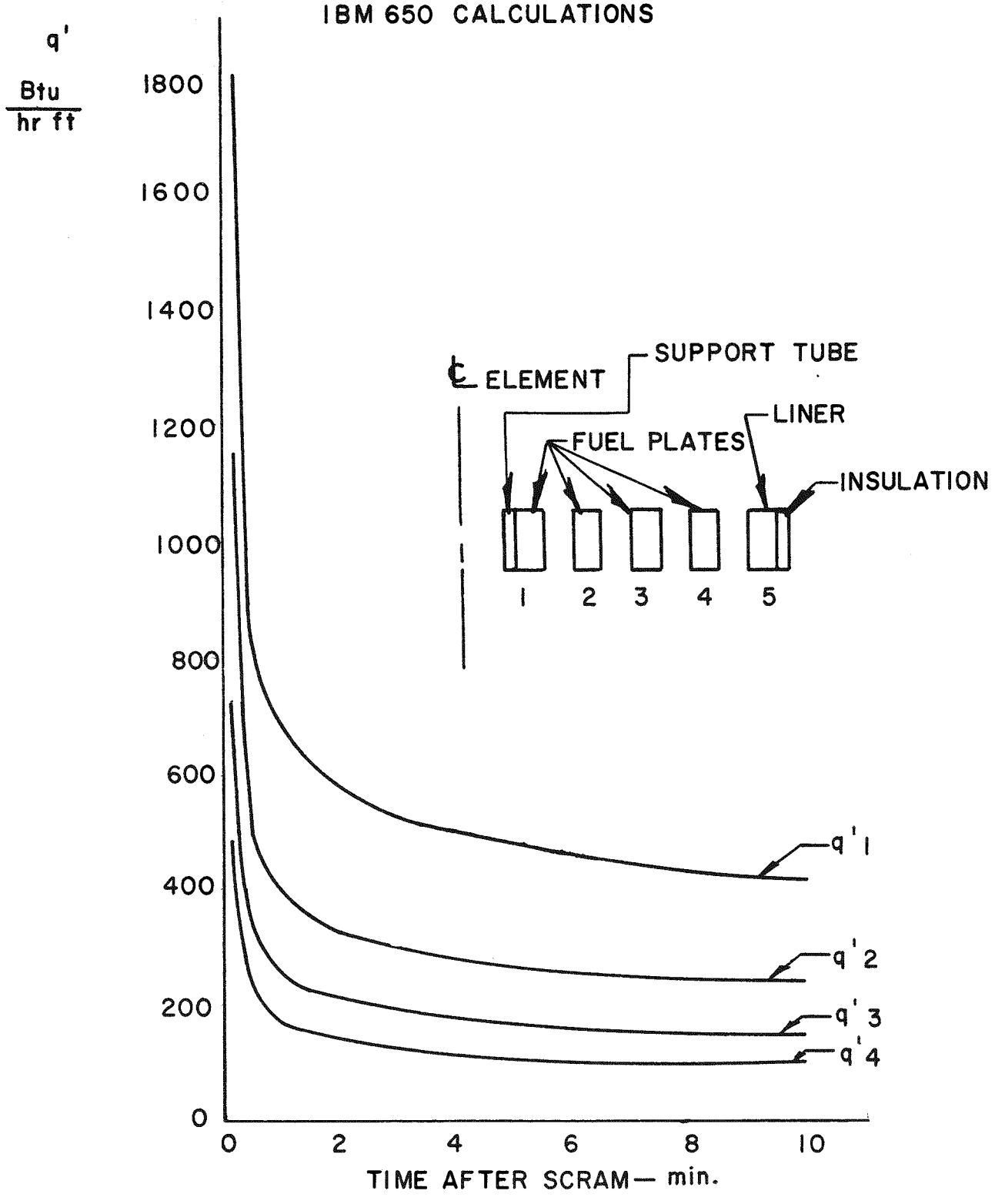
GCRE-I ELEMENT TEMPERATURES FOLLOWING COOLANT FLOW FAILURE, SCRAM AT I SEC.

FIG.-G-1

TRANSIENT GCRE HEAT GENERATION RATES

X/L = 0.5

IBM 650 CALCULATIONS



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FIG.-G-2

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